4. ENVIRONMENTAL CONSEQUENCES

Chapter 4 describes the environmental consequences of the proposed action and alternatives to treat and manage sodium-bonded spent nuclear fuel. It begins with a general discussion of the expected environmental consequences; the product and waste forms that would be generated from the proposed action; and the methodology for assessing health effects from radiological and chemical effluent. It follows with a detailed description of the environmental consequences for the No Action and the reasonable alternatives. The chapter provides separate discussions on the environmental consequences of the intersite transportation of sodium-bonded spent nuclear fuel; the cumulative impacts at each of the proposed sites; and the programmatic considerations associated with the proposed action. The chapter concludes with a look at several issues under the proposed action, such as unavoidable, adverse environmental impacts; relationships between local, short-term uses of the environment and the enhancement of long-term productivity; and irretrievable commitments of resources.

4.1 OVERVIEW OF ENVIRONMENTAL CONSEQUENCES

This *Final Environmental Impact Statement for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel* (SBSNF EIS) is in compliance with Council on Environmental Quality regulations that require the affected environment of proposed Federal actions to be "interpreted comprehensively to include the natural and physical environment and the relationship of people with the environment" (40 CFR 1508.14).

The environmental consequence analysis focused on potentially affected areas. These areas are discussed in detail: air quality, water resources, socioeconomics, public and occupational health and safety (normal operations and accident conditions), environmental justice, waste management, and transportation. For the remaining areas (i.e., land resources, visual resources, noise, geology and soils, ecological resources, and cultural and paleontological resources), analyses show that the proposed treatment activities would have minimal or no impact at the candidate sites regardless of the alternatives being considered. This is because existing facilities within developed areas would be used; no new land disturbance would take place and proposed activities would be consistent with current operations. Since none of the alternatives involve construction other than internal building modifications for installing new equipment, the effects of these modifications on any of the resources would be negligible and are not evaluated in this chapter.

The specific assumptions associated with the impact analysis common to all alternatives are provided in the appendices. The results of the assessment of environmental consequences are presented in this chapter. More detailed descriptions of the development of the impacts for some resource areas are presented in Appendices E through H, as follows:

- Appendix E, Evaluation of Human Health Effects From Normal Operations
- Appendix F, Evaluation of Human Health Effects From Facility Accidents
- Appendix G, Evaluation of Human Health Effects From Overland Transportation
- Appendix H, Environmental Justice Analysis

4.1.1 Presentation of the Environmental Consequences

The primary areas of concern are products and waste, impacts on the public, and occupational health and safety associated with the various sodium-bonded spent nuclear fuel treatment processes. Additional areas and topics covered in Chapter 4 include the following:

- Air Quality
- Water Resources
- Environmental Justice
- Socioeconomics
- Waste Management
- Transportation Impacts
- Short-term versus Long-term Resource Commitments
- Irreversible and Irretrievable Resource Commitments
- Cumulative Impacts

Several kinds of impacts are not discussed in Chapter 4 because they would not occur, they would be extremely small, and/or they are covered by other analyses:

Land Use—The treatment and management of sodium-bonded spent nuclear fuel would not require the construction of new facilities on previously undisturbed land at Argonne National Laboratory-West (ANL-W) or the Savannah River Site (SRS).

Intrasite Transportation—The incident-free impacts of intrasite transportation would be limited to radiation exposure to workers loading and unloading trucks and are included in the overall worker dose values presented for each process. The accident risks are bounded by the site accident risk analysis. Strict site safety procedures and short travel distances would limit the impacts to workers.

Noise—Noise impacts at the management sites would be minor and would be limited to noise generated during operations. No offsite noise impacts are expected except for minor changes in traffic noise levels.

Ecological Resources—Because no new construction in undisturbed areas would be required for the treatment and management of sodium-bonded fuel, there would be no disturbance to terrestrial and aquatic habitats or wetlands. Thus, there would be no negative impacts from construction on terrestrial or aquatic plants or animals, including threatened and endangered species.

Scientific evidence indicates that limiting chronic radiation doses to below 0.1 rad per day to the more radiosensitive species in terrestrial ecosystems provides adequate protection for the population. In the aquatic environment, limiting the chronic dose rate to 1 rad or less per day to an individual member of the aquatic population would provide adequate protection for the population (IAEA 1992). Therefore, limiting chronic radiation doses to below 0.1 rad per day would not harm animal or plant populations. This is equivalent to a dose of 100 millirem per day from direct radiation (inhalation dose) to an individual. Compliance with U.S. Department of Energy (DOE) Order 5400.5 to limit the exposure of the most exposed member of the public (a hypothetical individual residing at the site boundary) to 100 millirem per year (i.e., about 0.3 millirem per day from all pathways) and to 10 millirem per year from the air pathway (40 CFR 61 dose limit) makes it highly probable that dose rates to plants and animals in the same area would be less than 0.1 rad per day. The maximum annual dose to the most exposed member of the public under any one of the alternatives analyzed would be a small fraction (about 0.2 percent) of 1 millirem. Therefore, no radiological damage to plant and animal populations would be expected to result from the sodium-bonded spent nuclear fuel treatment processes.

Chemicals emitted to the environment during routine processing activities are presented under each alternative. These releases are essentially independent of the process being performed. They are generated from operation of support facilities, such as operation of emergency diesel generators during testing and/or fuel burning for facility heating and power production. The quantities of releases attributable to the treatment of the sodium-bonded spent nuclear fuel would be very small fractions of the current releases from each management site.

The site environmental reports did not identify any measurable impacts on plants or animals because the amounts emitted are very low or the chemicals have little potential for causing negative effects.

For the reasons discussed above, no adverse impacts to ecological resources would be expected to occur due to DOE's treatment and management of sodium-bonded spent nuclear fuel.

Cultural and Paleontological Resources—No new facilities would be needed or constructed, therefore, there would be no impacts on cultural or paleontological resources.

Geology and Soils—No new facilities would be needed or constructed. Therefore, there would be no disturbance to either geologic or soil resources at the management sites. Hazards from large-scale geologic conditions were analyzed in detail in various DOE programmatic environmental impact statements (EISs) and site-specific facility safety analysis reports. The impacts from these hazards (e.g., earthquakes) on the management facilities and treatment processes are evaluated in this EIS.

4.1.2 Products and Waste

Generation and Disposition—All of the treatment processing alternatives presented in this EIS, except for direct disposal in high-integrity cans, would change the sodium-bonded spent nuclear fuel into other forms. Driver and blanket sodium-bonded spent nuclear fuel would be input—products and waste would be the output. The products and waste would be better suited for storage, transportation, and disposal or other disposition than the existing sodium-bonded fuel. The products and waste fall into several distinct categories:

- Materials to be managed as high-level radioactive waste would be generated at SRS and/or ANL-W. The final form would be ceramic, metallic, a melt and dilute product, or borosilicate glass inside stainless steel canisters. The production of ceramic, metallic, and melt and dilute products at ANL-W would result mainly from the transformation of spent nuclear fuel to a different form that would make the final product more stable and lead to an overall reduction in repository volume need. This waste would be stored at SRS and/or ANL-W until a geologic repository is ready to receive it.

- Transuranic waste refers to processed materials that contain alpha-emitting material (such as plutonium) with radioactivity concentrations above 100 nanocuries per gram of waste. Transuranic waste would be generated from all treatment technologies. This waste could be disposed of in the Waste Isolation Pilot Plant.

- The separated uranium resulting from the electrometallurgical treatment process at ANL-W would be made into solid metal ingots. The separated uranium resulting from processing the driver spent nuclear fuel would be made into low-enriched uranium ingots. The ingots would be more than 99.7 percent pure uranium; the balance of the material would be mainly zirconium (the alloy in the fuel) and trace quantities of fission products and actinides requiring additional purification before the uranium ingots could be used commercially. The uranium ingots would be stored in secure facilities along with other uranium already in storage at ANL-W until decisions are made about their disposition.

- Separated depleted uranium from plutonium-uranium extraction (PUREX) processing of declad and cleaned blanket spent nuclear fuel at SRS would be made into uranium oxides and stored in drums along with other depleted uranium at SRS (more than 27,000 metric tons of depleted uranium are stored currently at SRS). The 57 metric tons of depleted uranium that would be separated in this process would be a small fraction of what is stored currently.

- Separated plutonium resulting from PUREX processing of declad and cleaned blanket spent nuclear fuel at F-Canyon would be in a metallic form. The separated plutonium, less than 260 kilograms (572 pounds),

would be stored in secure facilities along with the plutonium already in storage (about 2.4 metric tons) at SRS. The plutonium would be disposed of in accordance with the Record of Decision (75 *FR* 1608) for the *Surplus Plutonium Disposition Final Environmental Impact Statement* (DOE 1999d).

- Low-level radioactive waste would be generated from all treatment technologies considered. This waste would be disposed of in existing facilities using routine procedures.

- Saltstone would be generated only at SRS. Saltstone is a form of concrete containing low levels of radioactivity and would be disposed of on site.

- Mixed waste would be generated at ANL-W and SRS. At ANL-W, mixed waste would be generated mainly from cadmium contamination, which would be present in one of the electrorefiners. It would consist of waste categorized as indirect process solid waste and would include discarded equipment and materials from decontamination operations. At SRS, liquid mixed waste would be generated from contamination by various chemicals in the dissolution and extraction facilities.

Waste Minimization—DOE would incorporate the best available practices into all the processing technologies at the two management sites to generate the smallest possible amount of waste. The DOE sites managing the sodium-bonded spent nuclear fuel would comply with DOE's waste minimization and pollution prevention goals. The following summarizes recent achievements in pollution prevention and waste minimization at ANL-W and SRS:

- ANL-W conducted pollution prevention projects in 1997 that reduced waste generation by an estimated 1,700 cubic meters (61,100 cubic feet) at a cost savings of $154,000. Radioactive waste generation in 1997 was reduced by 61 percent compared to 1993 baseline levels. Mixed waste generation was increased by 67 percent, hazardous waste generation was reduced by 44 percent, and sanitary waste generation was reduced by 32 percent compared to baseline levels. Fifty-six percent of sanitary waste was recycled in 1997. ANL-W affirmative procurement purchases are not tracked separately, and are included in the Idaho National Engineering and Environmental Laboratory (INEEL) totals. For INEEL, 72 percent of the materials purchased were U.S. Environmental Protection Agency (EPA)-designated recycled products (DOE 1998e).

- SRS conducted pollution prevention projects in 1997 that reduced waste generation by an estimated 18,200 cubic meters (644,000 cubic feet) at a cost savings of $18.5 million. Radioactive waste generation in 1997 was reduced by 57 percent compared to 1993 baseline levels. Mixed waste generation was increased by 115 percent, hazardous waste generation was reduced by 15 percent, and sanitary waste generation was reduced by 58 percent compared to baseline levels. Seventy-eight percent of sanitary waste was recycled in 1997, and 52 percent of the materials purchased under the affirmative procurement process were EPA-designated recycled products (DOE 1998e).

4.1.3 Radiological and Chemical Health Risk Estimates

The methodologies used to evaluate potential radiological and chemical health effects from operational effluent are described in Appendix E. This section provides information about the development and interpretation of the health risk estimates.

Radiological—The effect of radiation on people depends upon the kind of radiation exposure (alpha, beta, and neutron particles and gamma and x-rays), duration of exposure, and the total amount of tissue exposed to radiation. The amount of radiant energy imparted to tissue from exposure to ionizing radiation is referred to as "absorbed dose." The sum of the absorbed dose to each tissue, when multiplied by certain quality and

weighting factors that take into account radiation quality and different sensitivities of these various tissues, is referred to as "effective dose equivalent."

An individual may be exposed to radiation from outside or inside the body, because radioactive materials may enter the body by ingestion or inhalation. External dose is different from internal dose in that it is delivered only during the actual time of exposure. An internal dose, however, continues to be delivered as long as the radioactive source is in the body (although both radioactive decay and elimination of the radionuclide by ordinary metabolic processes decrease the dose rate with the passage of time). The dose from internal exposure is calculated over 50 years following the initial exposure.

The regulatory annual radiation dose limits to the maximally exposed offsite individual from total operations at a DOE site are 10 millirem from atmospheric pathways, 4 millirem from drinking water pathways, and 100 millirem from all pathways combined (DOE Order 5400.5 and 40 CFR Part 61, Subpart H). The potential doses associated with the normal operation of various treatment technologies and storage of sodium-bonded spent nuclear fuel would be very small fractions of these values, and total site doses would remain well within these DOE limits. For comparison, DOE estimates that the average individual in the United States receives a dose of approximately 360 millirem per year from all radiation sources combined, including natural and medical sources (see Appendix E, Section E.2.1, for details).

The collective or "population" dose to an exposed population is calculated by summing the estimated doses received by each member of the exposed population. The collective dose received by the exposed population is measured in person-rem. For example, if 1,000 people each received a dose of 0.001 rem, the population dose would be 1 person-rem (1,000 persons × 0.001 rem = 1 person-rem). The same population dose (1 person-rem) would result if 500 people each received a dose of 0.002 rem (500 persons × 0.002 rem = 1 person-rem).

Radiation can cause a variety of adverse health effects in people. A large dose of radiation can cause prompt death. At low doses of radiation, the most important adverse health effect from environmental and occupational radiation exposures (which are typically low doses) is the potential inducement of fatal cancers. This effect is referred to as "latent" cancer fatalities because the cancer may take years to develop and for death to occur.

In addition to latent cancer fatalities, other health effects could result from exposures to radiation. These effects include nonfatal cancers among the exposed population and genetic effects in subsequent generations. The dose-to-effect factors for fatal and nonfatal cancers are shown in **Table 4–1**. As indicated in this table, the nonfatal cancers and genetic effects are less probable consequences per unit of radiation exposure. For simplicity, this EIS presents estimated effects of radiation only in terms of latent cancer fatalities. Estimates of the total detriment (fatal cancers, nonfatal cancers, and genetic effects) due to radiation exposure may be obtained from the estimates of latent cancer fatalities presented in this EIS by multiplying by 1.4 for workers and by 1.46 for the general public.

The factors used in this EIS to relate a dose to its effect are 0.0004 lifetime probability of a latent cancer fatality per person-rem for workers and 0.0005 lifetime probability of a latent cancer fatality per person-rem for individuals among the general population. The latter factor is slightly higher because some individuals in the public, such as infants and children, are more sensitive to radiation than workers. These factors are based on the 1990 Recommendations of the International Commission on Radiological Protection (ICRP 1991) and are consistent with those used by the U.S. Nuclear Regulatory Commission (NRC) in its rulemaking *Standards for Protection Against Radiation* (10 CFR 20). The factors apply where the dose to an individual is less than 20 rem and the dose rate is less than 10 rem per hour. At higher doses and dose rates, the factors used to relate radiation doses to latent cancer fatalities are doubled. At much higher doses, prompt effects, rather than latent cancer fatalities risk, may be the primary concern.

Table 4–1 Risk of Latent Cancer Fatalities and Other Health Effects From Exposure to 1 Rem of Radiation[a]

Individual [b]	*Latent Cancer Fatalities*	*Nonfatal Cancers*	*Genetic Effects*	*Total Detriment*
Worker	0.0004	0.00008	0.00008	0.00056
Public	0.0005	0.0001	0.00013	0.00073

[a] When applied to an individual, units are lifetime probability of a latent cancer fatality per rem of radiation dose. When applied to a population of individuals, units are the excess number of cancers per person-rem of radiation dose. Genetic effects as used here apply to populations, not individuals.

[b] The difference between the worker risk and the general public risk is attributable to the fact that the general population includes more individuals in the more sensitive age group of less than 18 years of age.

Note: One rem equals 1,000 millirem.

Sources: NCRP 1993, ICRP 1991.

These factors are used to calculate the statistical expectations of the effects of exposing a population to radiation. For example, if 100,000 people each were exposed to a one-time radiation dose of 100 millirem (0.1 rem), the collective dose would be 10,000 person-rem. The exposed population then would be expected to experience 5 additional latent cancer fatalities from the radiation (10,000 person-rem × 0.0005 lifetime probability of a latent cancer fatality per person-rem = 5 latent cancer fatalities).

Sometimes calculations of the number of latent cancer fatalities associated with radiation exposure do not yield whole numbers and, especially in environmental applications, may yield numbers less than 1. For example, if 100,000 people each were exposed to a total dose of only 1 millirem (0.001 rem), the collective dose would be 100 person-rem, and the corresponding estimated number of excess latent cancer fatalities would be 0.05 (100,000 persons × 0.001 rem × 0.0005 latent cancer fatalities per person-rem = 0.05 latent cancer fatalities). The "0.05" means that there is one chance in 20 that the exposed population would experience one latent fatal cancer. In other words, the latent cancer fatality rate of 0.05 is the *expected* number of deaths that would result if the same exposure situation were applied to many different groups of 100,000 people. In most groups, nobody (0 people) would incur a latent cancer fatality from the 1 millirem dose each member would have received. In a small fraction of the groups, one latent cancer fatality would result; in exceptionally few groups, two or more latent cancer fatalities would occur. The average expected number of deaths for all the groups would be 0.05 latent cancer fatalities (just as the average of 0, 0, 0, and 1 is 1/4, or 0.25). The most likely outcome is 0 latent cancer fatalities.

The same concept is applied to estimate the effects of continuous radiation exposure to an individual member of the public. Consider the effects of an individual's exposure to a 360-millirem (0.36-rem) annual dose from all radiation sources (natural and medical). The probability that the individual would develop a latent fatal cancer from continuous exposure to this radiation over an average life of 72 years (presumed) is 0.013 (1 person × 0.36 rem per year × 72 years × 0.0005 latent cancer fatality risk per person rem = 0.013), or one chance in 77 that the individual would develop a fatal cancer from this radiation exposure.

The estimates of health effects from radiation doses used in this EIS are based on the linear no-threshold theory of radiation carcinogenesis, which postulates that all radiation doses, even those close to 0, are harmful. As explained in Appendix E, the numerical estimates of fatal cancers presented in this EIS were obtained from the nominal risk estimated for lifetime total cancer mortality, resulting from a dose of 0.1 gray (10 rad) (National Research Council 1990). Other methods of extrapolation to the low-dose region could yield higher or lower numerical estimates of fatal cancers. Studies of human populations exposed to low doses are inadequate to demonstrate the actual level of risk. There is a scientific uncertainty about cancer risk in the low-dose region below the range of epidemiologic observation, and the possibility of no risk cannot be excluded (CIRRPC 1992).

This EIS provides radiation dose estimates and probabilities of latent cancer fatalities (risks) for various receptors from management facility radiation exposure during normal operations and accident conditions. The receptors are defined as follows:

Worker – An individual actively participating and/or supporting the operation of the facility.

Noninvolved worker– An individual who is not involved in the operation of the facility. For estimating the impact, the individual is assumed to be 100 or more meters (330 or more feet) from the radioactive or chemical material release point.

Maximally exposed offsite individual – An individual member of the public assumed to be residing at the site boundary who could receive the maximum dose from radiation or hazardous chemicals.

Population – members of the general public residing within an 80-kilometer (50-mile) radius of the facility.

For incident-free (normal) operations, the EIS provides two sets of impacts (dose and risk): maximum annual and project total impacts for all alternatives. The maximum annual impacts result from simultaneous treatment of both driver and blanket sodium-bonded spent nuclear fuel in a given year, and the project total impacts represent the overall impacts from treatment of all sodium-bonded spent nuclear fuel. For accident conditions, the EIS provides both the consequence (dose) per accident and the associated risk.

Chemical—The potential impacts of exposure to hazardous chemicals released to the atmosphere as a result of the processing of sodium-bonded spent nuclear fuel were evaluated for incident-free operations and accident conditions at management facilities. Small amounts of hazardous and toxic chemical releases would be expected from incident-free operation of the treatment technology support facilities and equipment (e.g., auxiliary steam power house, diesel generators). The health effects from these releases were calculated for the maximally exposed offsite individual (an individual member of the public residing at the site boundary). The health effects evaluated in this analysis include excess latent cancer fatalities and chemical-specific noncancer health effects. The maximally exposed offsite individual was assumed to be located in the region with the highest estimated concentration. The health effects from releases of hazardous chemicals during accident conditions were evaluated in terms of comparison to Emergency Response Planning Guideline (ERPG) values. ERPG values are estimates of airborne concentration thresholds above which one can reasonably anticipate to observe adverse effects (see Appendix F, Section F.3.1.2, for more detail).

4.2 NO ACTION ALTERNATIVE

Under the No Action Alternative, the sodium-bonded spent nuclear fuel would not be treated (no sodium would be removed from the fuel). Under this alternative, two options were considered.

The EIS evaluates:

a. The impacts from the activities required to monitor and stabilize the sodium-bonded spent nuclear fuel as necessary for continued safe and secure storage at current locations, or until a new treatment technology, such as the glass material oxidation and dissolution system (GMODS) or plasma arc, is developed (see Section 2.6 for more details on GMODS and plasma arc technology development needs).

b. The impacts from direct disposal of sodium-bonded spent nuclear fuel in a geologic repository by packaging the fuel in high-integrity cans without sodium removal. At the present time, direct disposal of sodium-bonded spent nuclear fuel is precluded by DOE policy concerning acceptance of Resource Conservation and Recovery Act (RCRA)-designated mixed waste (which contains both hazardous and radioactive waste).

Under either option of the No Action Alternative, the EIS evaluates the impacts associated with activities required to clean and stabilize the waste materials generated during the Electrometallurgical Treatment Research and Demonstration Project at ANL-W. As part of this demonstration project, approximately 1.6 metric tons of heavy metal of Experimental Breeder Reactor -II (EBR-II) fuel consisting of about 1.2 metric tons of blanket spent nuclear fuel and 0.4 metric tons of driver spent nuclear fuel were processed (DOE 1996b). The waste materials generated in this project currently are being transformed into ceramic and metallic waste forms. This process will continue until all of this waste is transferred to ceramic and metallic waste forms. The remaining sodium-bonded spent nuclear fuel in the treatment facilities will be packaged and transferred to dry storage in the Radioactive Scrap and Waste Facility.

DOE also is transferring to dry storage all INEEL spent nuclear fuel, including the sodium-bonded spent nuclear fuel currently stored at Idaho Nuclear Technology and Engineering Center (INTEC) Building 603 (wet storage basin). During this transfer, each fuel can containing sodium-bonded fuel will be nondestructively examined to determine the fuel can condition and its suitability for storage. If any fuel can is found to be degraded, resulting in water in-leakage, it will be repackaged and transferred to ANL-W for stabilization and/or repackaging for storage. The fuel transfer activities are planned for completion by December 2000. The sodium-bonded spent nuclear fuel currently stored at INTEC Building 666 (wet storage basin) will remain in the basin until the planned defueling and facility closure in the year 2023. These fuel movement activities would be performed independently of the activities within this EIS.

About 5 metric tons of heavy metal of EBR-II blanket spent nuclear fuel contained in 107 storage cans currently stored at the Radioactive Scrap and Waste Facility do not meet the long-term confinement requirements. Under the No Action Alternative continued safe storage option. These storage cans would be brought to the Hot Fuel Examination Facility to be repackaged in more durable storage liners and would be returned to storage. These activities, along with the waste processing activities at ANL-W, would be completed in about two years after the necessary waste handling equipment is installed. The sodium-bonded spent nuclear fuel that currently is stored at INTEC would remain there, and sodium-bonded spent nuclear fuel transferred to Idaho in the future, as specified in the amended Record of Decision for the *Department of Energy Programmatic Spent Nuclear Fuel Management and Idaho Engineering Laboratory Environmental Restoration and Waste Management Programs Final Environmental Impact Statement* (Programmatic Spent Nuclear Fuel EIS) (61 FR 9441), also would be stored at INTEC. Consistent with the DOE-State of Idaho Settlement Agreement and Consent Order, all spent nuclear fuel would need to be transferred out of the State of Idaho by January 1, 2035. Under this option of the No Action Alternative, in the event that sodium-bonded spent nuclear fuel has not been treated before 2035, DOE would package the stored fuel at ANL-W and transfer it to the INEEL Dry Transfer facility. DOE also may decide to use the facilities at ANL-W to package the sodium-bonded spent nuclear fuel stored at INTEC. In the event that the sodium-bonded spent nuclear fuel has not been treated before 2035, the stored fuel would be removed from the State of Idaho by the year 2035. The environmental impacts of untreated sodium-bonded spent nuclear fuel removal would be evaluated in a separate National Environmental Policy Act (NEPA) document.

Under the No Action Alternative direct disposal option, all sodium-bonded spent nuclear fuel at INTEC would be transferred to ANL-W and repackaged in high-integrity cans in preparation for direct disposal. The activities associated with the preparation of sodium-bonded spent nuclear fuel for direct disposal would be similar to those needed to prepare the fuel for continued safe storage. The activities for direct disposal would occur sometime after those for the continued storage option. This is because a decision to directly dispose of the

sodium-bonded spent nuclear fuel in a geologic repository would be made only after it was determined that it would meet the repository acceptance criteria. Currently, there are no acceptance criteria for this fuel type. If direct disposal of the sodium-bonded spent nuclear fuel becomes possible, DOE would use the facilities at ANL-W to prepare all sodium-bonded spent nuclear fuel at the INEEL site. Preparation of driver spent nuclear fuel for direct disposal requires consideration of criticality safety, thereby limiting the amount of driver spent nuclear fuel that could be packaged in a canister. This would lead to larger repository volume needs per unit mass for driver fuel.

The activities in this option would include:

1. Repackaging 107 cans containing 5 metric tons of heavy metal of blanket spent nuclear fuel in the first two years (ending in 2003); see the continued storage option above.

2. Transferring sodium-bonded spent nuclear fuel currently stored at INTEC (Building 666 Basin and Building 603 Dry Storage) to ANL-W between 2003 and 2023. The 2023 date corresponds to the target date for closure of Building 666 Basin at INTEC. Under this assumption, the fuel in Building 666 Basin would be in wet storage for 23 years.

3. Repackaging the spent nuclear fuel at ANL-W in high-integrity cans to meet the target date for fuel transfer out of the State of Idaho (January 1, 2035). All sodium-bonded spent nuclear fuel would be transferred to the Hot Fuel Examination Facility for characterization and placement in high-integrity cans. The preparation and canning activities would be completed in about three years. The canned fuel would be stored temporarily at the Radioactive Scrap and Waste Facility. The stored fuel cans would be packaged in standardized canisters and transferred to the INEEL Dry Transfer Facility for packaging and shipment to the repository.

The environmental impacts for both options under the No Action Alternative are presented below. Where the impacts are different between the options, two sets of results are presented.

4.2.1 Air Quality

Nonradiological Gaseous Emissions

As explained in Appendix E, Section E.5.3.1, under the proposed action and either option of the No Action Alternative, small quantities of criteria pollutants and hazardous chemicals are generated from the operation of the emergency diesel generators supporting both the Fuel Conditioning Facility and the Hot Fuel Examination Facility at ANL-W. The emissions from these generators are independent of any of the treatment processes under the proposed action and the No Action Alternative addressed in this EIS.

Table 4–2 summarizes the concentrations of criteria and hazardous air pollutants. The concentrations are compared to their corresponding ambient air quality standards. Only those air pollutants that are expected and have ambient air quality standards are presented in the table. The emissions are generated from diesel generators currently in operation and are considered as part of the baseline concentration. No increases in emissions are expected under the No Action Alternative. Therefore, a Prevention of Significant Deterioration increment analysis was not required. In addition, the INEEL site is located in areas of attainment for the criteria pollutants; therefore, no conformity analysis is required.

Table 4–2 Nonradiological Air Quality Concentrations at the Site Boundary Under the No Action Alternative for Comparison With Ambient Air Quality Standards

	Averaging Period	*Most Stringent Standard or Guideline (micrograms per cubic meter)* [a]	*Maximum Incremental Concentration (micrograms per cubic meter)*
Criteria Pollutant			
Carbon monoxide	8 hours 1 hour	10,000 40,000	32.7 46.8
Nitrogen dioxide	Annual	100	2.8
PM_{10}	Annual 24 hours (interim) 24 hours (99^{th} percentile over 3 years)	50 150 150	0.01 0.19 Not available
$PM_{2.5}$	3-year annual 24 hours (98^{th} percentile over 3 years)	15 65	Not available Not available
Sulfur dioxide	Annual 24 hours 3 hours	80 365 1,300	0.45 11.50 25.80
Hazardous and Toxic Compounds			
1,3-Butadiene	Annual	0.0036	0.0000355
Acetaldehyde	Annual	0.45	0.0000226
Acrolein	24 hours	12.5	0.000181
Benzene	Annual	0.12	0.000694
Formaldehyde	Annual	0.077	0.0000709
Toluene	24 hours	18,750	0.00664
Xylene	24 hours	21,750	0.00447

PM_n = Particulate matter less than or equal to *n* microns in diameter.

[a] The standards for hazardous and toxic compounds apply only to increases in emissions from new or modified sources and are provided for information purposes only, as the concentrations from releases at ANL-W under all alternatives are not expected to increase.

Radiological Gaseous Emissions

Potential radiological releases from sodium-bonded spent nuclear fuel would be very small under both options of this alternative. Under both options, the spent nuclear fuel would remain stored in sealed canisters while at INEEL (i.e., INTEC or ANL-W) until 2035. However, degradation of sodium-bonded spent nuclear fuel or its enclosure (e.g., a sealed canister) during storage cannot be ruled out. It is expected that a small fraction of the fuel would degrade during storage, allowing its gaseous fission products to enter the storage canister. These fission gases would be released to the environment only if the sealed canister were to fail or be opened during fuel handling for examination and repackaging. As detailed in Appendix E, Section E.4.6, current experience at INTEC and ANL-W indicates very small fuel degradation problems during the storage period. It was estimated that, over 30 years of storage, about 1 percent of the fuel would be in degraded condition while in dry storage and about 3 percent of the fuel would fail while in wet storage. While in dry storage, there would be no releases of gaseous fission products to the environment. The fission gases would be released to the environment only during fuel repackaging. In wet storage, fuel canister degradation and resulting fuel failure would lead to releases of gaseous fission products. The estimated gaseous fission product releases during the entire period (over 35 years) of the No Action Alternative would be 51 curies of tritium oxides, 760 curies of krypton-85, and 0.000018 curies of iodine-129.

4.2.2 Water Resources

Surface Water

No surface water is used at ANL-W. Flood waters from the Big Lost River would not be expected to reach the facilities at ANL-W, as shown in Figure 3–3.

Nonradiological Liquid Effluent

There are no discharges to the surface waters at ANL-W, except for discharges of nonhazardous liquid waste to the sewage pond and to the industrial waste pond. Discharge waters to the Industrial Waste Pond or to the Sanitary Sewage Lagoons are not waters of the United States and are exempt from compliance under the National Pollutant Discharge Elimination System (NPDES). However, these are designated as waters of the State of Idaho and, as such, require compliance with State regulations that govern application of nonhazardous liquid waste (i.e., Land Application Permits). ANL-W has applied to the State of Idaho for Land Application Permits for the Industrial Waste Pond and Ditches and the Sanitary Waste Treatment Pond Land Application Area (DOE 1996a, DOE 1998c). ANL-W routinely monitors the effluent discharges to make sure they are within those limits identified in the Land Application Permits. Current operating and monitoring practices would continue for stormwater and liquid effluent discharges associated with facilities at ANL-W.

Radiological Liquid Effluent

No radiological liquid effluent would be discharged to the surface water.

Groundwater

Under either option of this alternative, there would be some reduction in groundwater consumption for domestic uses, since the number of workers at ANL-W is expected to decrease. The current water use at ANL-W is 188 million liters (49.6 million gallons) per year.

Nonradiological Liquid Effluent

For either option of this alternative, no nonradiological liquid effluent or waste would be discharged to groundwater.

Radiological Liquid Effluent

For either option of this alternative, no radiological liquid effluent would be discharged to groundwater.

4.2.3 Socioeconomics

Under either option of the No Action Alternative, there could be a reduction of approximately 350 workers at ANL-W if a treatment technology is not selected or the decision is delayed. If all of these workers were to leave the regional economic area, this could result in the loss of an additional 940 indirect jobs in the economic region. The total potential loss of about 1,290 jobs represents less than a 1 percent decrease in civilian employment in the regional economic area, which was estimated to be 150,403 in 1996 (DOE 1999d).

Since any reduction in the ANL-W labor force under the No Action Alternative would take place over time, combined with the fact that many of these workers could also support other missions at INEEL, the effects are expected to be gradual. By 2010, the contributory effect of this and the potential for beneficial effects

from other industrial and economic sectors within the regional economic area would serve to reduce or mask any effect on the regional economy. Neither option of the No Action Alternative, therefore, would result in any noticeable change in the existing regional economy, population and housing characteristics, or community services within the region of influence at ANL-W (see Section 3.2.8).

4.2.4 Public and Occupational Health and Safety

The assessments of potential radiological and chemical impacts associated with the No Action Alternative are presented in this section. Summaries of radiological impacts from normal operations are presented in Tables 4–3 and 4–4 for the public and workers, respectively. The radiological impacts from a spectrum of hypothetical accident scenarios are provided in Tables 4–6 and 4–7. The impacts from hazardous chemical releases during accident conditions are presented in Table 4–8. Background information on the effects of radiation on human health and safety is presented in Section 4.1.3 and Appendix E, Section E.2.

4.2.4.1 Normal Operations

Radiological Impacts

Under either option of the No Action Alternative, radioactive releases from normal operations associated with spent nuclear fuel storage activities at ANL-W and INTEC would be small. The releases would occur from fuel degradation in wet storage and during fuel handling. Under both options, the same amount of gaseous radioactive material would be released. As explained in Appendix E, Section E.4.6, under both options, some fuel would be repackaged at the beginning in the first two years and all of the fuel would be repackaged by 2035 prior to shipment outside the INEEL site. The repackaging would occur over a three-year period. Releases would occur both from INTEC during wet storage and from ANL-W during fuel handling and repackaging operations. However, since INTEC is further away from the INEEL site boundary and major population centers compared to ANL-W, the releases were assumed to occur from ANL-W, thereby maximizing the impacts.

Calculated maximum annual and project total radiological impacts to the public are given in **Table 4–3**. The impacts are calculated for two types of receptors: the general public living within 80 kilometers (50 miles) of ANL-W in the year 2010, and a maximally exposed offsite individual (a member of the public assumed to be residing at the INEEL site boundary and receiving the maximum dose). To put the operational impacts into perspective, comparisons with impacts from natural background radiation also are included in the table. As shown in this table, the expected radiation doses to the maximally exposed offsite individual and the general public would be much smaller than the limit of 10 millirem per year limit set by the EPA (40 CFR 61) and DOE (DOE Order 5400.5).

The average worker dose (for ANL-W and INTEC workers) under the No Action Alternative was estimated to be similar to that currently experienced at ANL-W. Under both options, waste and fuel handling and repackaging activities would occur over a 5-year period, with standby operations for the remaining 30 years. One additional year also would be necessary to deactivate the facility. During fuel handling operations, the estimated annual total worker population dose would be 22 person-rem; during storage (standby) operations, it would be 2.2 person-rem; and during deactivation, it would be 33 person-rem, for a total of 209 person-rem over 35 years (see **Table 4–4**).

Table 4–3 Annual and Project Total Radiological Impacts to the Public From Operational Activities Under the No Action Alternative

Receptor	*Impacts*
Population Within 80 Kilometers (50 Miles) in the Year 2010	
Collective dose (person-rem per year) [a]	0.0015
Excess latent cancer fatalities (per year)	7.5×10^{-7}
Project total excess latent cancer fatalities [b]	6.5×10^{-6}
Maximally Exposed Offsite Individual	
Dose (millirem per year) [a]	0.00026
Percent of annual background [c]	0.000072
Latent cancer fatality risk (per year)	1.3×10^{-10}
Project total lifetime cancer fatality risk [b]	1.1×10^{-9}
Average Individual Within 80 Kilometers (50 Miles)	
Dose (millirem per year) [d]	6.2×10^{-6}
Latent cancer fatality risk	3.1×10^{-12}
Project total lifetime cancer fatality risk [b]	2.7×10^{-11}

[a] Annual maximum dose during normal operations.
[b] Total calculated risk over 35 years.
[c] The annual natural background radiation level at INEEL is about 360 millirem for the average individual (see Table 3–8); the population within 80 kilometers (50 miles) in the year 2010 would receive 86,500 person-rem.
[d] Obtained by dividing the population dose by the number of people projected to live within 80 kilometers (50 miles) of ANL-W in the year 2010 (240,338).

Table 4–4 Annual and Project Total Radiological Impacts to Workers From Operational Activities Under the No Action Alternative

Receptor	*Impacts*
Worker [a]	
Average worker dose (millirem per year)	60
Average worker latent cancer fatality risk (project total over 35 years)	0.00084
Worker Population	
Collective dose (person-rem per year)	22
Excess latent cancer fatalities (per year)	0.0088
Project total dose (person-rem)	209
Project total excess latent cancer fatalities	0.084

[a] The regulatory dose limit for an individual worker is 5,000 millirem per year (10 CFR 835). However, the maximum annual dose to a worker would be kept below the DOE Administrative Control Level of 2,000 millirem per year, as established for all DOE activities in DOE Order N 441.1.

Source: ANL 1999.

As shown in Tables 4–3 and 4–4:

- The maximum annual dose to the maximally exposed offsite individual would be 0.00026 millirem, with an associated risk of developing a lifetime fatal cancer of 1.3×10^{-10} per year (or one chance in 7.7 billion that the individual would develop a fatal cancer per year of exposure).

- The collective maximum annual dose to the population within 80 kilometers (50 miles) of the storage facilities at ANL-W would be 0.0015 person-rem, with an associated 7.5×10^{-7} latent cancer fatalities per year (or one chance in 1.3 million that the population would experience a fatal cancer per year of exposure).

- The project total risk to the population within 80 kilometers (50 miles) of ANL-W would be 6.5×10^{-6} latent cancer fatalities (or one chance in 154,000 that the exposed population would experience a fatal cancer).

- The collective dose to facility workers would be 22 person-rem per year, with an associated 0.0088 latent cancer fatalities per year (or one chance in 113 that the workers would experience a fatal cancer per year of operation).

- The project total dose to facility workers would be 209 person-rem with an associated 0.084 latent fatal cancers (or one chance in 12 that the exposed workers would experience a fatal cancer).

These results indicate that no increase in the expected number of fatal cancers among the various affected populations would result from the activities under this alterative.

Hazardous Chemical Impacts

The hazardous chemical impacts to the public residing at the INEEL site boundary from the operational activities at ANL-W under either option of this alternative are summarized in **Table 4–5**. Appendix E, Section E.5, provides details on the model used and results obtained. The results (presented in Table 4–5) indicate that no adverse toxic health or cancer effects would be expected from exposure to hazardous chemicals released under this alternative. The existing chemical environment is presented in Section 3.2.10.2.

Table 4–5 Hazardous Chemical Impacts to the Public From Operational Activities Under the No Action Alternative

Chemical	*Maximum Annual Concentration (milligrams per cubic meter)*	*Hazard Quotient (noncarcinogenic chemicals)*	*Cancer Risk (carcinogenic chemicals)*
1,3-Butadiene	3.6×10^{-8}	None	9.9×10^{-9}
Acetaldehyde	2.3×10^{-8}	2.5×10^{-6}	5.0×10^{-11}
Acrolein	7.1×10^{-9}	0.00035	None
Benzene	6.9×10^{-7}	None	5.4×10^{-9}
Formaldehyde	7.1×10^{-8}	None	9.2×10^{-10}
Toluene	2.5×10^{-7}	6.2×10^{-7}	None
Hazard Index		0.00036	Not applicable

4.2.4.2 Facility Accidents

The potential radiological impacts to the public and noninvolved onsite workers due to accidents are summarized in this section. The detailed analysis of facility accidents, with the associated assumptions, is presented in Appendix F. The detailed analysis considered a wide spectrum of potential accident scenarios, including fire, spills, criticality, an earthquake, and an aircraft crash.

Under either option of the No Action Alternative, spent nuclear fuel transfer and waste processing activities associated with cleaning and stabilizing the waste materials generated during the Electrometallurgic

Treatment Research and Demonstration Project at ANL-W would be performed. These activities would have the potential to involve accident scenarios similar to those evaluated for Alternative 1 as presented in Section 4.3.4.2. However, the consequences associated with these accident scenarios would be lower because of the limited quantities of waste to be stabilized. Accidents associated with spent nuclear fuel transfer activities also could occur during fuel removal from the Radioactive Scrap and Waste Facility and packaging for offsite shipment to a repository. These accidents would lead to consequences similar to those evaluated for Alternative 1 as presented in Section 4.3.4.2. It is estimated that the spent nuclear fuel transfer and waste stabilization activities would occur over a two-year period. Fuel handling and repackaging for offsite shipment would occur over a three-year period.

No reasonably foreseeable accident scenarios could be identified that would impact sodium-bonded spent nuclear fuel in dry storage at the Radioactive Scrap and Waste Facility or in wet or dry storage at INTEC. In storage, the sodium-bonded spent nuclear fuel is in a safe and stable configuration. Generally, the only activity associated with the stored spent nuclear fuel is monitoring of the fuel and the storage facility. While in storage, activities that could lead to accidents (movement, repackaging, or processing of the spent nuclear fuel) are not performed. However, approximately 1.2 metric tons of sodium-bonded spent nuclear fuel currently in wet storage in Basin 603 at INTEC would be transferred to dry storage facilities at INTEC. Handling accidents could occur during transfer activities at INTEC similar to the accident scenarios evaluated for ANL-W. Because INTEC is further away from the INEEL site boundary and major population centers compared to ANL-W, the health impacts from accidents at INTEC would be less than those from similar accidents at ANL-W.

Table 4–6 presents the frequencies and consequences of the postulated set of accidents to the maximally exposed offsite individual, the offsite population residing within 80 kilometers (50 miles) of the facility, and a noninvolved worker. The dose to the maximally exposed offsite individual was calculated for the 95th percentile meteorological conditions. The doses to the population and the noninvolved worker were calculated using 50th percentile meteorological conditions. The 50th percentile condition represents the median meteorological condition, and is defined as that for which more severe conditions occur 50 percent of the time. The 95th percentile condition represents relatively low probability meteorological conditions that produce higher calculated exposures, and is defined as that condition that is not exceeded more than 5 percent of the time. DOE did not quantitatively estimate the involved worker dose due to accidents. The consequences to involved workers are qualitatively assessed. This approach is used for the following two reasons: (1) no adequate method exists for calculating meaningful consequences at or near the location where the accident occurs, and (2) safety assurance for facility workers is demonstrated by both the workers' training and by the establishment of an Occupational Safety and Health Administration (OSHA) process safety management system (29 CFR 1910.119). In any accident scenario, the individuals most likely to be injured are the involved workers. The risk to these workers would be due to both radiological and nonradiological effects. In a fire, the involved workers could be exposed to airborne radioactive material, in addition to the smoke and heat of the fire. In an explosion, there could be flying debris and containment barriers could be broken, exposing workers to airborne radioactive material. Most spills would not have a major effect on involved workers because they would clean up the spill wearing protective clothing and respirators as necessary. An accidental criticality could expose involved workers to large doses of prompt penetrating radiation, which could cause death in a short period of time. An earthquake accident could present a very severe nonradiological effect to the involved workers. For example, in a beyond-design-basis earthquake, the workers are likely to be hurt or could be killed from the collapse of the building before they could be evacuated (see Appendix F, Section F.2.2.2, for more detail). The accident risks are summarized in **Table 4–7**.

Table 4–6 Accident Frequency and Consequences Under the No Action Alternative

Accident [a]	Frequency (event per year)	Maximally Exposed Offsite Individual		Population Within 80 Kilometers (50 Miles)		Noninvolved Worker	
		Dose (millirem)	Latent Cancer Fatality Risk [b]	Dose (person-rem)	Excess Latent Cancer Fatalities [c]	Dose (millirem)	Latent Cancer Fatality Risk [b]
Salt powder spill in the Hot Fuel Examination Facility cell [d]	0.01	0.00046	2.3×10^{-10}	0.000098	4.9×10^{-8}	4.7×10^{-7}	1.9×10^{-13}
Cask drop during spent nuclear fuel transfer	0.01	0.03	1.5×10^{-8}	0.0035	1.7×10^{-6}	0.00084	3.4×10^{-10}
Transuranic waste fire	0.001	0.059	3.0×10^{-8}	0.0071	3.6×10^{-6}	0.22	8.8×10^{-8}
Design-basis earthquake	0.008	12	6.0×10^{-6}	1.4	0.00070	4.7	1.9×10^{-6}
Salt transfer drop	1×10^{-7}	0.19	9.5×10^{-8}	0.022	0.000011	0.073	2.9×10^{-8}
Beyond-design-basis earthquake	0.00001	96	0.000048	11	0.0055	37	0.000015

[a] Only accidents involving EBR-II driver spent nuclear fuel, which maximizes the consequences, are presented.
[b] Increased likelihood of a latent cancer fatality.
[c] Increased number of latent cancer fatalities.
[d] The salt powder spill was assumed to have similar characteristics to those evaluated under Alternative 1. The radionuclide concentration in this salt would be about one-third of those generated in Alternative 1.

Table 4–7 Annual Cancer Risks Due to Accidents Under the No Action Alternative

Accident	Maximally Exposed Offsite Individual [a]	Population Within 80 Kilometers (50 Miles) [b]	Noninvolved Worker [a]
Salt powder spill in Hot Fuel Examination Facility cell	2.3×10^{-12}	4.9×10^{-10}	1.9×10^{-15}
Cask drop during spent nuclear fuel transfer	1.5×10^{-10}	1.7×10^{-8}	3.4×10^{-12}
Transuranic waste fire	3.0×10^{-11}	3.6×10^{-9}	8.8×10^{-11}
Design-basis earthquake	4.8×10^{-8}	5.6×10^{-6}	1.5×10^{-8}
Salt transfer drop	9.5×10^{-15}	1.1×10^{-12}	2.9×10^{-15}
Beyond-design-basis earthquake	4.8×10^{-10}	5.5×10^{-8}	1.5×10^{-10}

[a] Increased likelihood of a latent cancer fatality.
[b] Increased number of latent cancer fatalities.

For accidents at ANL-W, the highest risk of a latent cancer fatality to the maximally exposed offsite individual and to a noninvolved worker would be 4.8×10^{-8} per year (or one chance in 20.8 million that the individual would develop a fatal cancer per year of operation) and 1.5×10^{-8} per year (or one chance in 66.7 million that the individual would develop a fatal cancer per year of operation), respectively. The increased number of latent cancer fatalities in the surrounding population would be 5.6×10^{-6} per year (or one chance in 178,600 that the population would experience a fatal cancer per year of operation).

Hazardous Chemical Impacts

Nonradiological hazardous chemical impacts are evaluated in terms of comparison to ERPGs. ERPG values are estimates of airborne concentration thresholds above which one can reasonably anticipate observing adverse effects (see Appendix F, Section F.3.1.2, for details).

The nonradiological (hazardous chemical) impacts of potential facility accidents associated with either option of the No Action Alternative are summarized in **Table 4–8**.

Table 4–8 Hazardous Chemical Accident Impacts Under the No Action Alternative

Accident	*Frequency (Event Per Year)*	*Receptor*	*Exposure*
Uranium handling accident	0.01	Noninvolved worker	Uranium: less than ERPG-1
		Maximally exposed offsite individual	Uranium: less than ERPG-1
Design-basis earthquake	0.0002	Noninvolved worker	Uranium: less than ERPG-1
		Maximally exposed offsite individual	Uranium: less than ERPG-1
Beyond-design-basis earthquake	0.00001	Noninvolved worker	Cadmium: less than ERPG-1 Uranium: less than ERPG-1
		Maximally exposed offsite individual	Cadmium: less than ERPG-1 Uranium: less than ERPG-1

ERPG = Emergency Response Planning Guideline.

4.2.5 Environmental Justice

As discussed in Appendix H, Executive Order 12898 directs Federal agencies to address disproportionately high and adverse health or environmental effects of alternatives on minority or low-income populations.

Analyses of normal operations and accident conditions presented in Section 4.2.4 show the risk of latent cancer fatalities to the public residing within 80 kilometers (50 miles) of the spent nuclear fuel storage facilities at ANL-W and INTEC to be much lower than 1. Therefore, there would be no disproportionately high and adverse consequences for any particular group within the general population, including minority or low-income populations, beyond the effects of existing and future activities that are independent of the proposed action.

4.2.6 Waste Management

Various types of waste would be generated as a result of sodium-bonded spent nuclear fuel storage activities at ANL-W, including transuranic waste, low-level radioactive waste, mixed waste, hazardous, and nonhazardous waste. In addition, during the first two years of operation under either option of this alternative, ANL-W would continue to generate high-level radioactive waste as the Electrometallurgical Treatment Research and Demonstration Project cladding hull waste and electrorefiner salt are stabilized to metallic and ceramic high-level radioactive waste forms for ultimate disposal. **Table 4–9** shows the anticipated categorization of these waste types and their expected interim storage and final disposal locations. The quantities of ceramic and metallic waste forms generated, along with other generated waste, are presented in **Table 4–10**. The values in Table 4–10 are for disposal (solid waste) and account for volume reduction.

Direct Process Waste

Under either option of the No Action Alternative, small amounts of metallic and ceramic high-level radioactive waste would be produced at ANL-W as a result of the completion of the demonstration project. The salt removed from the electrorefiners would contain the majority of fission products and transuranics from the spent nuclear fuel. This removed salt would be packaged and transferred to the Hot Fuel Examination Facility for processing into ceramic waste. The metallic waste form would consist primarily of stainless steel cladding hulls containing the noble metal fission products. Both the ceramic and metallic waste would be categorized as high-level radioactive waste. The volumes of waste forms provided in Table 4–10 are for the standardized canisters required for disposal of these materials.

Table 4–9 Waste Material Categories at INEEL and Interim and Final Locations

Waste Stream	*Category*	*Interim Storage Location*	*Final Disposal Location*
Process Waste			
Fuel hardware	Low-level radioactive waste	None	Radioactive Waste Management Complex
Metallic waste form	High-level radioactive waste	Radioactive Scrap and Waste Facility	Geologic repository
Ceramic waste form	High-level radioactive waste	Radioactive Scrap and Waste Facility	Geologic repository
Other Associated Process Waste			
Less than 10 nanocuries per gram transuranic waste [a]	Low-level radioactive waste	None	Radioactive Waste Management Complex
Greater than 100 nanocuries per gram transuranic waste	Transuranic waste	Radioactive Waste Management Complex	Waste Isolation Pilot Plant
Cadmium-contaminated	Mixed waste	Radioactive Scrap and Waste Facility	Waste Isolation Pilot Plant or Radioactive Waste Management Complex after treatment
Nonradioactive	Sanitary waste	None	INEEL landfill
Deactivation Waste			
Electrorefiner cadmium	Mixed waste	Radioactive Scrap and Waste Facility	Waste Isolation Pilot Plant
Equipment less than 10 nanocuries per gram transuranic waste [a]	Low-level radioactive waste	None	Radioactive Waste Management Complex
Equipment greater than 100 nanocuries per gram transuranic waste	Transuranic waste	Radioactive Waste Management Complex	Waste Isolation Pilot Plant
Cadmium-contaminated	Mixed waste	Radioactive Scrap and Waste Facility	Waste Isolation Pilot Plant or Radioactive Waste Management Complex

[a] As noted in Section 3.2.11.3, the Radioactive Waste Management Complex cannot be used for the disposal of the alpha low-level radioactive waste (between 10 and 100 nanocuries per gram). Waste in this category may be treated by the Advanced Mixed Waste Treatment Project and then disposed of at the Waste Isolation Pilot Plant.

The metallic and ceramic high-level radioactive waste generated as a result of the demonstration project at ANL-W would be stored temporarily at the Radioactive Scrap and Waste Facility at ANL-W in a manner that allows retrieval for future disposal. The Radioactive Scrap and Waste Facility was designed and constructed for temporary storage of this type of waste, and shielding will be provided by a combination of: (1) steel storage liners in which the waste would be stored and (2) by the soil surrounding the liners. When a geologic repository is available, the waste cans containing the metallic and ceramic high-level radioactive waste would be removed from storage, packaged in standardized canisters, shipped to the INEEL Dry Transfer Facility, and prepared for shipment to the repository. If direct disposal of sodium-bonded spent nuclear fuel becomes acceptable, the sodium-bonded spent nuclear fuel at INTEC would be transferred to ANL-W for repackaging, along with other fuel at ANL-W. The packaged canisters would be transferred to the INEEL Dry Transfer Facility for shipment off site to a repository.

Table 4–10 Amounts of Waste Generated Under the No Action Alternative[a]

Waste Stream	Waste Quantities	
	Volume (cubic meters)	Mass (kilograms)
Direct Process Waste		
Fuel assembly hardware (low-level radioactive waste)	0	0
High-level radioactive ceramic waste	9.4 (15 canisters) [b]	14,000
High-level radioactive metallic waste	0.6 (1 canister) [b]	460
Spent nuclear fuel	142 (355 canisters) [b]	72,000
Other Associated Process Waste		
Low-level radioactive waste	792	161,000
Transuranic waste	10.5	4,000
Mixed waste	40	21,500
Sanitary waste	2,500	867,000
Deactivation Waste		
Low-level radioactive waste	112	38,000
Transuranic waste	1.6	853
Mixed waste	3	2,100

[a] These waste generation estimates are through the year 2035. This is the date by which materials of this type are required to be out of the State of Idaho.
[b] Standardized canisters.
Source: ANL 1999.

Other Associated Process Low-Level Radioactive Waste

Low-level radioactive waste would be generated during conversion of demonstration high-level radioactive waste into suitable forms for the repository, as well as from other ongoing activities, including keeping a hot cell facility operational to handle unforeseen problems while storing the sodium-bonded spent nuclear fuel at INTEC or in the Radioactive Scrap and Waste Facility. In addition, low-level radioactive waste would be generated from activities in the Fuel Conditioning Facility and the Hot Fuel Examination Facility (e.g., equipment decontamination and repair), as well as in other facilities at ANL-W (e.g., analytical laboratory activities). Material in this waste stream has been generated and routinely handled at ANL-W for many years.

The volume of low-level radioactive waste resulting from either option of the No Action Alternative activities at ANL-W that would require disposal (after volume reduction) would be a maximum of about 50 cubic meters (1,766 cubic feet) per year during processing activities, and approximately 17 cubic meters (600 cubic feet) per year during the remaining years. This maximum volume represents a small fraction (approximately 1 percent) of the total annual volume of low-level radioactive waste currently being disposed of at the Radioactive Waste Management Complex. The total of 792 cubic meters (28,000 cubic feet) of low-level radioactive waste generated during either option represents approximately 0.7 percent of the total Radioactive Waste Management Complex disposal inventory.

Other Associated Process Transuranic Waste

Transuranic waste would be generated at ANL-W under either option of the No Action Alternative from decontamination activities for repair and maintenance of items, and miscellaneous work associated with

demonstration fuel processing or other activities. Transuranic waste would be generated primarily from activities conducted in gloveboxes and hot cells at ANL-W.

For the No Action Alternative, the volume of transuranic waste generated at ANL-W would amount to a maximum of approximately 1 cubic meter (35 cubic feet) per year during processing activities, and approximately 0.2 cubic meters (7 cubic feet) per year during the remaining years. This maximum volume is approximately 0.002 percent of the volume of transuranic waste in retrievable storage at the Radioactive Waste Management Complex. The total volume of incidental transuranic waste generated under either option is approximately 10.5 cubic meters (370 cubic feet), which is 0.006 percent of the estimated total volume of transuranic waste to be emplaced at the Waste Isolation Pilot Plant.

Other Associated Process Sanitary Waste

Sanitary waste, which is nonradioactive and nonhazardous solid waste, would continue to be generated under either option of the No Action Alternative. This waste would be typical of industrial operations and would be disposed of at the INEEL landfill. Based on an estimated eventual INEEL landfill volume of 3×10^6 cubic meters (106 million cubic feet), the total volume of sanitary waste generated and disposed of under this alternative is approximately 0.1 percent of the INEEL landfill volume.

Other Associated Process Mixed Waste

Mixed waste would be generated primarily from the disposal of any cadmium-contaminated equipment or clean-up material and the analysis of cadmium samples. At ANL-W, mixed waste would be handled according to ANL-W procedures that require limited accumulation at the point of generation. Interim storage of this waste would be at the Radioactive Scrap and Waste Facility prior to eventual disposal. The Radioactive Scrap and Waste Facility is a permitted mixed waste storage facility for these materials. The mixed waste streams that contribute to the overall mixed waste generated at ANL-W have been identified in the INEEL Site Treatment Plan (DOE 1995b).

Deactivation Waste

A variety of waste would be generated as part of deactivation activities at ANL-W. This would include process equipment and process material such as cadmium in one of the electrorefiners. Generated waste categories would include low-level radioactive waste, transuranic waste, and mixed waste. This waste would be categorized and disposed of according to DOE Orders and ANL-W radioactive waste management procedures, as described above for each waste category.

The largest volume of deactivation waste under either option of the No Action Alternative would be low-level radioactive waste, generated as a result of equipment dismantling and disposal. Components that would require disposal include the existing electrorefiners and hot isostatic press, as well as other processing components. Decontamination of these components would generate additional mixed, transuranic, and low-level radioactive waste that would require management. The deactivation waste volume would be generated over a period of one year. The total deactivation waste represents approximately 14 percent over the total associated process waste (excluding sanitary waste) requiring disposal.

4.3 ALTERNATIVE 1: ELECTROMETALLURGICALLY TREAT BLANKET AND DRIVER FUEL AT ANL-W

Under this alternative, the sodium-bonded spent nuclear fuel would be treated at ANL-W using the electrometallurgical process, described in Appendix C. The various process steps in this technology are performed at the Fuel Conditioning Facility and the Hot Fuel Examination Facility hot (air or argon) cells. The

processes at the Fuel Conditioning Facility include: fuel chopping, electrorefining, cathode processing, and metal casting (see Appendix C for details on each processing step). These processes would separate the uranium from the fission products. Separated uranium is not considered a waste. The separated uranium would be made into low-enriched uranium ingots, and the metallic sodium would be oxidized in the electrorefiner lithium-potassium salt and removed along with the fission products as high-level radioactive waste. The salts from the electrorefiner then would be solidified and sent to the Hot Fuel Examination Facility for further processing. The processes at the Hot Fuel Examination Facility include waste treatment, metallic melting, and high-level radioactive waste production. These processes would produce two waste forms—a ceramic waste form consisting of fission products and transuranic elements and a metallic waste form consisting of noble metal fission products and cladding hulls from the spent nuclear fuel. The low-enriched uranium metal ingots would be stored at the Zero Power Physics Reactor Material Storage Building. The ceramic and metallic waste forms would be temporarily stored at the Radioactive Scrap and Waste Facility pending packaging for disposition in a geologic repository.

The electrometallurgical process at ANL-W facilities would treat about 5 metric tons of heavy metal of sodium-bonded spent nuclear fuel per year. Appendix E, Section E.4.1, provides details on the process duration and the amount of blanket and driver spent nuclear fuel treated annually. The treatment of blanket and driver spent nuclear fuel under this alternative could start as early as 2000 and could be completed by 2012. In addition, a full year of operations would be needed to deactivate the processing equipment and establish an industrially safe configuration.

4.3.1 Air Quality

Nonradiological Gaseous Emissions

As explained in Appendix E, Section E.5.3.1, under all alternatives, small quantities of criteria and hazardous chemicals are generated from the operation of the emergency diesel generators supporting both the Fuel Conditioning Facility and the Hot Fuel Examination Facility at ANL-W. The emissions from these generators are independent of any of the treatment processes addressed in this EIS. In addition, the electrometallurgical treatment of driver fuel under Alternatives 1 through 5 would release small quantities of cadmium. This release would occur as an elevated release from the Fuel Conditioning Facility stack.

Table 4–11 summarizes the concentrations of criteria and hazardous air pollutants. The concentrations are compared to their corresponding ambient air quality standards. Only those air pollutants that are expected and have ambient air quality standards are presented in the table. The emissions are generated from diesel generators currently in operation and are considered as part of the baseline concentration. No increases in emissions are expected under this alternative. Therefore, a Prevention of Significant Deterioration increment analysis was not required. In addition, the INEEL site is located in areas of attainment for criteria pollutants; therefore, no conformity analysis is required.

Radiological Gaseous Emissions

Krypton-85 and elemental tritium are the most prevalent radioactive gaseous fission products that would be released to the argon cell at the Fuel Conditioning Facility during fuel element chopping and electrorefining processes. The released tritium in the cell would not be oxidized due to a very low presence of oxygen and humidity in the argon cell, and releases of tritium to the atmosphere would be in the elemental form. The oxidation of elemental tritium to tritium oxide (HTO or T_2O) has been shown to occur slowly in the environment, and in the long term, about 1 percent of tritium would be oxidized (see Appendix E, Section E.4.1, for more details). The argon cell also contains an equilibrium concentration of other radionuclide isotopes. Appendix E, Section E.4.1, provides a list of various isotopes that are present in the argon cell in nanocuries

(10^{-9} curies) and are released to the environment through the facility stack, along with krypton and elemental tritium. The maximum release of radioactive gaseous emissions would occur during the first six years of the electrometallurgical treatment process, when a combination of EBR-II blanket and driver spent nuclear fuel elements would be processed. During these six years, about 0.6 metric tons of heavy metal of driver spent nuclear fuel and about 4.4 metric tons of heavy metal of blanket spent nuclear fuel would be processed annually. The combined process would release about 11,600 curies of krypton-85 and 770 curies of elemental tritium annually. After six years and until the end of the processing period, the release rate would drop significantly. During this period, only Fermi-1 blanket spent nuclear fuel, with an annual release of about 0.4 curies of elemental tritium and 3.3 curies of krypton-85, would be processed. The radiological exposures to the public and workers from these emissions are presented in detail in Appendix E, Section E.4.1, and are summarized in Section 4.3.4.

Table 4–11 Nonradiological Air Quality Concentrations at the Site Boundary Under Alternative 1 at ANL-W for Comparison With Ambient Air Quality Standards

	Averaging Period	*Most Stringent Standard or Guideline (micrograms per cubic meter)* [a]	*Maximum Incremental Concentration (micrograms per cubic meter)*
Criteria Pollutant			
Carbon monoxide	8 hours 1 hour	10,000 40,000	32.70 46.80
Nitrogen dioxide	Annual	100	2.79
PM_{10}	Annual 24 hours (interim) 24 hours (99^{th} percentile over 3 years)	50 150 150	0.01 0.19 Not available
$PM_{2.5}$	3-year annual 24 hours (98^{th} percentile over 3 years)	15 65	Not available Not available
Sulfur dioxide	Annual 24 hours 3 hours	80 365 1,300	0.45 11.50 25.80
Hazardous and Toxic Compounds			
1,3-Butadiene	Annual	0.0036	0.0000355
Acetaldehyde	Annual	0.45	0.0000226
Acrolein	24 hours	12.5	0.000181
Benzene	Annual	0.12	0.000694
Cadmium	Annual	0.00056	3.58×10^{-10}
Formaldehyde	Annual	0.077	0.0000709
Toluene	24 hours	18,750	0.00664
Xylene	24 hours	21,750	0.00447

PM_n = Particulate matter less than or equal to *n* microns in diameter.

[a] The standards for hazardous and toxic compounds apply only to increases in emissions from new or modified sources and are provided for information purposes only, as concentrations from releases at ANL-W under all alternatives are not expected to increase.

4.3.2 Water Resources

Surface Water

No surface water is used at ANL-W. Flood waters from the Big Lost River would not be expected to reach the facilities at ANL-W, as shown in Figure 3–3.

Nonradiological Liquid Effluent

There are no discharges to the surface waters at ANL-W, except for discharges of nonhazardous liquid waste to the sewage pond and to the industrial waste pond. Big Lost River, Little Lost River, and Birch Creek would not be impacted by activities associated with the electrometallurgical treatment processes. Current operating and monitoring practices would continue for stormwater and liquid effluent discharges associated with facilities at ANL-W (see Section 4.2.2).

During fuel treatment and associated activities, some hazardous materials may be used inside buildings. To prevent potential releases to surface or subsurface waters resulting from spills of hazardous materials used in buildings, these facilities are designed, constructed, and maintained to contain these materials. Double-contained pipes, leak detection, and secondary containment of tanks are some of the features used to prevent hazardous material releases to the environment. Following existing written procedures, spill containment and cleanup equipment is present in areas where hazardous materials are stored or used (DOE 1996b).

Radiological Liquid Effluent

No radiological liquid effluent or waste generated by the electrometallurgical treatment process would be discharged to surface water.

Groundwater

Under this alternative at ANL-W, there would be little change in groundwater consumption for domestic use since there is little change expected in the number of workers. The current water use at ANL-W is 188 million liters (49.6 million gallons) per year.

Nonradiological Liquid Effluent

No nonradiological liquid effluent generated by the electrometallurgical treatment process would be discharged to groundwater.

Radiological Liquid Effluent

No radiological liquid effluent generated by the electrometallurgical treatment process would be discharged to groundwater.

4.3.3 Socioeconomics

Under this alternative, the existing facilities at ANL-W would remain operational. No new employment or in-migration of workers would be required. Thus, there would be no additional impacts on the socioeconomic conditions in the region around ANL-W and INEEL.

4.3.4 Public and Occupational Health and Safety

The assessments of potential incremental radiological and chemical impacts associated with this alternative are presented in this section. Summaries of radiological impacts from normal operations are presented in Tables 4–12 and 4–13 for the public and workers, respectively. The radiological impacts from a spectrum of hypothetical accident scenarios are provided in Tables 4–15 and 4–16. The impacts from hazardous chemical releases during accident conditions are presented in Table 4–17. Background information on the effects of radiation on human health and safety is presented in Appendix E, Section E.2.

4.3.4.1 Normal Operations

Radiological Impacts

Under this alternative, radioactive releases would occur during fuel chopping and from the operation of electrorefiners. Both of these activities are performed in the Fuel Conditioning Facility argon cell. Appendix E, Sections E.3 and E.4.1, details the method and assumptions used for calculating the impacts of normal operational radiological releases on the public health and safety. The maximum annual dose to the public would result from treating 0.6 metric tons of heavy metal of EBR-II driver spent nuclear fuel and 4.4 metric tons of heavy metal of EBR-II blanket spent nuclear fuel. This combination of fuel treatment would continue for 6 years, after which only Fermi-1 blanket spent nuclear fuel with a very low radioactivity content would be treated. Overall, it would require 13 years to treat all the sodium-bonded fuel (see Appendix E, Section E.4.1, for details).

Calculated maximum annual and project total radiological impacts to the public from operational activities under this alternative are given in **Table 4–12**. The maximum dose to the public would occur during the first six years of operation. The annual dose to the public during Fermi-1 blanket spent nuclear fuel treatment would be very small (see Appendix E, Table E–8). The impacts are calculated for two types of receptors: the general public living within 80 kilometers (50 miles) of ANL-W in the year 2010, and a maximally exposed offsite individual (a member of the public assumed to be residing at the INEEL site boundary and receiving the maximum dose). As explained in Appendix E, Section E.4.1, the dose resulting from the release of tritium depends heavily on the chemical form. The inhalation dose from oxidized tritium (HTO or T_2O) is 25,000 times higher than for elemental tritium (HT or T_2). In the environment, about 1 percent of elemental tritium would be oxidized over the long term. In this analysis, about 1 percent of the tritium conservatively was assumed to be in oxidized form at the time of release.

Table 4–12 Annual and Project Total Radiological Impacts to the Public From Operational Activities Under Alternative 1

Receptor	*Electrometallurgically Treat Driver Spent Nuclear Fuel*	*Electrometallurgically Treat Blanket Spent Nuclear Fuel*	*Total*
Population Within 80 Kilometers (50 Miles) in the Year 2010			
Collective dose (person-rem per year) [a]	0.0027	0.000083	0.0028
Excess latent cancer fatalities (per year)	1.4×10^{-6}	4.2×10^{-8}	1.4×10^{-6}
Project total excess latent cancer fatalities [b]	8.0×10^{-6}	2.2×10^{-7}	8.2×10^{-6}
Maximally Exposed Offsite Individual			
Dose (millirem per year) [a]	0.00033	0.000010	0.00034
Percent of annual background radiation [c]	0.000092	2.8×10^{-6}	0.000094
Latent cancer fatality risk (per year)	1.6×10^{-10}	5.0×10^{-12}	1.7×10^{-10}
Project total lifetime cancer fatality risk [b]	9.6×10^{-10}	2.6×10^{-11}	9.9×10^{-10}
Average Individual Within 80 Kilometers (50 Miles)			
Dose (millirem per year) [d]	0.000011	3.5×10^{-7}	0.000012
Latent cancer fatality risk (per year)	5.6×10^{-12}	1.7×10^{-13}	5.8×10^{-12}
Project total lifetime cancer fatality risk [b]	3.3×10^{-11}	9.1×10^{-13}	3.4×10^{-11}

[a] Annual maximum dose during normal operations.
[b] Total calculated risk over 13 years of emissions.
[c] The annual natural background radiation level at INEEL is about 360 millirem for the average individual (see Table 3–8); the population within 80 kilometers (50 miles) in the year 2010 would receive 86,500 person-rem.
[d] Obtained by dividing the population dose by the number of people projected to live within 80 kilometers (50 miles) of ANL-W in the year 2010 (240,338).

Primary contributors to doses to members of the public are from releases of tritium gas and krypton-85, which together contribute over 99.9 percent of the total calculated doses. To put the operational impacts into perspective, comparisons with impacts from natural background radiation also are included in Table 4–12. As shown in the table, the expected radiation doses to the maximally exposed offsite individual and the general public are much smaller than the limit of 10 millirem per year set by the EPA (40 CFR 61) and DOE (DOE Order 5400.5).

Occupational doses were estimated by examining the type and duration of various operations performed by ANL-W workers involved with the electrometallurgical treatment of sodium-bonded spent nuclear fuel. The estimated annual worker population collective dose would be 22 person-rem, with an average individual dose of 60 millirem per year for each of the 346 involved workers. If these estimates were extended over the 13 years of electrometallurgical treatment activities, and a 1-year dose (33 person-rem) from deactivation activities is included, the project total worker population dose would be 319 person-rem, leading to a risk of 0.13 latent cancer fatalities (see **Table 4–13**).

Table 4–13 Annual and Project Total Radiological Impacts to Workers From Operational Activities Under Alternative 1

Receptor	*Impacts*
Worker [a]	
Average worker dose (millirem per year)	60
Average worker latent cancer fatality risk project total over 13 years	0.00031
Worker Population	
Collective dose (person-rem per year)	22
Excess latent cancer fatalities (per year)	0.0088
Project total dose (person-rem)	319
Project total excess latent cancer fatalities	0.13

[a] The regulatory dose limit for an individual worker is 5,000 millirem per year (10 CFR 835). However, the maximum annual dose to a worker would be kept below the DOE Administrative Control Level of 2,000 millirem per year as established for all DOE activities in DOE Order N 441.1.

Source: ANL 1999.

As shown in Tables 4–12 and 4–13:

- The maximum dose to the maximally exposed offsite individual would be 0.00034 millirem per year, with an associated risk of developing a fatal cancer of 1.7×10^{-10} per year (or one chance in 5.9 billion that the individual would develop a fatal cancer per year of exposure).

- The maximum collective dose to the population within 80 kilometers (50 miles) of the ANL-W facilities would be 0.0028 person-rem per year, with an associated 1.4×10^{-6} latent cancer fatalities per year (or one chance in 667,000 that the population would experience a fatal cancer per year of exposure).

- The project total risk to the population within 80 kilometers (50 miles) of the ANL-W facilities would be 8.2×10^{-6} latent cancer fatalities (or one chance in 122,000 that the exposed population would experience a fatal cancer).

- The collective dose to facility workers would be 22 person-rem per year, with an associated 0.0088 latent cancer fatalities per year (or one chance in 113 that the workers would experience a fatal cancer per year of operation).

- The project total dose to facility workers would be 319 person-rem, with an associated 0.13 latent cancer fatalities (or one chance in seven that the workers would experience a fatal cancer).

These results indicate that no increase in the expected number of fatal cancers among the various affected populations would result from the activities under this alternative.

Hazardous Chemical Impacts

The hazardous chemical impacts to the public residing at the INEEL site boundary from the operational activities at ANL-W under this alternative are summarized in **Table 4–14**. Appendix E, Section E.5, provides details on the model used and results obtained. The results, presented in Table 4–14, indicate that no adverse toxic health or cancer effects would be expected from exposure to hazardous chemicals released under this alternative. The existing chemical environment is presented in Section 3.2.10.2.

Table 4–14 Hazardous Chemical Impacts to the Public From Operational Activities Under Alternative 1

Chemical	*Maximum Annual Concentration (milligrams per cubic meter)*	*Hazard Quotient (noncarcinogenic chemicals)*	*Cancer Risk (carcinogenic chemicals)*
1,3-Butadiene	3.6×10^{-8}	None	9.9×10^{-9}
Acetaldehyde	2.3×10^{-8}	2.5×10^{-6}	5.0×10^{-11}
Acrolein	7.1×10^{-9}	0.00035	None
Benzene	6.9×10^{-7}	None	5.4×10^{-9}
Cadmium	3.6×10^{-13}	None	6.5×10^{-13}
Formaldehyde	7.1×10^{-8}	None	9.2×10^{-10}
Toluene	2.5×10^{-7}	6.2×10^{-7}	None
Hazard Index		0.00036	Not applicable

4.3.4.2 Facility Accidents

Radiological Impacts

Potential radiological impacts to the public and a noninvolved onsite worker due to accidents during electrometallurgical treatment operational activities are summarized and presented in this section. Since electrometallurgical treatment processes are performed in both the Fuel Conditioning Facility and the Hot Fuel Examination Facility, accidents at both facilities would be considered. The detailed analysis of facility accidents, with the associated assumptions, is presented in Appendix F. The detailed analysis considered a wide spectrum of potential accident scenarios including fire, spills, criticality, earthquake, and aircraft crash. Aircraft crash and criticality accidents were determined to have an accident frequency of less than 10^{-7} per year, and were not analyzed further. **Table 4–15** presents the frequencies and consequences of the postulated set of accidents to the maximally exposed offsite individual; the offsite population residing within 80 kilometers (50 miles) of the facility; and a noninvolved worker located 100 meters (330 feet) to 230 meters (755 feet) from the facility. The 230-meter (755-foot) distance is the ANL-W bus staging area, which leads to a higher dose to the noninvolved worker for the scenarios with elevated releases.

The dose to the maximally exposed offsite individual was calculated for 95th percentile meteorological conditions. The doses to the population and the noninvolved worker were calculated using 50th percentile meteorological conditions. DOE did not quantitatively estimate the involved worker dose due to accidents

(see discussions on the involved worker in Section 4.2.4.2). The accident risks are summarized in **Table 4–16**.

Table 4–15 Accident Frequency and Consequences Under Alternative 1

Accident	*Frequency (event per year)*	*Maximally Exposed Offsite Individual*		*Population Within 80 Kilometers (50 Miles)*		*Noninvolved Worker*	
		Dose (millirem)	*Latent Cancer Fatality Risk* [a]	*Dose (person-rem)*	*Excess Latent Cancer Fatalities* [b]	*Dose (millirem)*	*Latent Cancer Fatality Risk* [a]
Driver Spent Nuclear Fuel							
Salt powder spill	0.01	0.00046	2.3×10^{-10}	0.000098	4.9×10^{-8}	4.7×10^{-7}	1.9×10^{-13}
Salt transfer drop	1.0×10^{-7}	0.19	9.5×10^{-8}	0.022	0.000011	0.073	2.9×10^{-8}
Transuranic waste fire	0.001	0.059	3.0×10^{-8}	0.0071	3.6×10^{-6}	0.22	8.8×10^{-8}
Cask drop	0.01	0.030	1.5×10^{-8}	0.0035	1.8×10^{-6}	0.00084	3.4×10^{-10}
Design-basis earthquake	0.008	12	6.0×10^{-6}	1.4	0.0007	4.7	1.9×10^{-6}
Beyond-design-basis earthquake	0.00001	22,000	0.022	2,500	1.3	370	0.00015
Blanket Spent Nuclear Fuel							
Salt powder spill	0.01	0.00015	7.5×10^{-11}	0.000033	1.7×10^{-8}	1.3×10^{-6}	5.3×10^{-13}
Salt transfer drop	1.0×10^{-7}	0.065	3.3×10^{-8}	0.0077	3.9×10^{-6}	0.22	8.8×10^{-8}
Transuranic waste fire	0.001	0.059	3.0×10^{-8}	0.0071	3.6×10^{-6}	0.22	8.8×10^{-8}
Cask drop	0.01	0.0024	1.2×10^{-9}	0.00028	1.4×10^{-7}	0.000049	2.0×10^{-11}
Design-basis earthquake	0.008	4.0	2.0×10^{-6}	0.47	0.00024	14	5.6×10^{-6}
Beyond-design-basis earthquake	0.00001	930	0.00047	110	0.055	560	0.00023

[a] Increased likelihood of a latent cancer fatality.
[b] Increased number of latent cancer fatalities.

Table 4–16 Annual Cancer Risks Due to Accidents Under Alternative 1

Accident	*Maximally Exposed Offsite Individual* [a]	*Population Within 80 Kilometers (50 Miles)* [b]	*Noninvolved Worker* [a]
Driver Spent Nuclear Fuel			
Salt powder spill	2.3×10^{-12}	4.9×10^{-10}	1.9×10^{-15}
Salt transfer drop	9.5×10^{-15}	1.1×10^{-12}	2.9×10^{-15}
Transuranic waste fire	3.0×10^{-11}	3.6×10^{-9}	8.8×10^{-11}
Cask drop	1.5×10^{-10}	1.7×10^{-8}	3.4×10^{-12}
Design-basis earthquake	4.8×10^{-8}	5.6×10^{-6}	1.5×10^{-8}
Beyond-design-basis earthquake	2.2×10^{-7}	0.000013	1.5×10^{-9}
Blanket Spent Nuclear Fuel			
Salt powder spill	7.5×10^{-13}	1.7×10^{-10}	5.3×10^{-15}
Salt transfer drop	3.3×10^{-15}	3.9×10^{-13}	8.8×10^{-15}
Transuranic waste fire	3.0×10^{-11}	3.6×10^{-9}	8.8×10^{-11}
Cask drop	1.2×10^{-11}	1.4×10^{-9}	2.0×10^{-13}
Design-basis earthquake	1.6×10^{-8}	1.9×10^{-6}	4.5×10^{-8}
Beyond-design-basis earthquake	4.7×10^{-9}	5.5×10^{-7}	2.3×10^{-9}

[a] Increased likelihood of a latent cancer fatality.
[b] Increased number of latent cancer fatalities.

For accidents at ANL-W, the highest risk of a latent cancer fatality to the maximally exposed offsite individual and to a noninvolved worker would be 2.2×10^{-7} per year (or one chance in 4.5 million that the individual would develop a fatal cancer per year of operation) and 4.5×10^{-8} per year (or one chance in 22.2 million that the individual would develop a fatal cancer per year of operation), respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.000013 per year (or one chance in 76,920 that the population would experience a fatal cancer per year of operation).

Hazardous Chemical Impacts

Nonradiological impacts are evaluated in terms of comparison to ERPGs. ERPG values are estimates of airborne concentration thresholds above which one can reasonably anticipate observing adverse effects (see Appendix F, Section F.3.1.2, for details). The nonradiological impacts of potential facility accidents associated with the electrometallurgical treatment alternative at ANL-W are summarized in **Table 4–17**.

Table 4–17 Hazardous Chemical Accident Impacts Under Alternative 1

Accident	*Frequency (event per year)*	*Receptor*	*Exposure*
Uranium handling accident	0.01	Noninvolved worker	Uranium: less than ERPG-1
		Maximally exposed offsite individual	Uranium: less than ERPG-1
Design-basis earthquake	0.0002	Noninvolved worker	Uranium: less than ERPG-1
		Maximally exposed offsite individual	Uranium: less than ERPG-1
Uranium fire	0.00001	Noninvolved worker	Uranium: less than ERPG-1
		Maximally exposed offsite individual	Uranium: less than ERPG-1
Beyond-design-basis earthquake	0.00001	Noninvolved worker	Cadmium: less than ERPG-1
			Uranium: less than ERPG-1
		Maximally exposed offsite individual	Cadmium: less than ERPG-1
			Uranium: less than ERPG-1

ERPG = Emergency Response Planning Guideline.

4.3.5 Environmental Justice

As discussed in Appendix H, Executive Order 12898 directs Federal agencies to address disproportionately high and adverse health or environmental effects of alternatives on minority or low-income populations.

Analyses of normal operations and accident conditions presented in Section 4.3.4 show the risk of latent cancer fatalities to the public residing within 80 kilometers (50 miles) of the electrometallurgical processing facilities at ANL-W to be much lower than 1. Therefore, radiological and nonradiological risks posed by implementation of this alternative would have no disproportionately high and adverse consequences on any particular group within the general population, including minority or low-income populations.

4.3.6 Waste Management

Electrometallurgical treatment of sodium-bonded spent nuclear fuel at ANL-W would generate process waste from treatment operations, other associated process waste from normal support operations, and deactivation waste following the conclusion of operations. Process waste would include fuel hardware and high-level radioactive metallic and ceramic waste. Other associated process waste would include operational waste such as failed equipment, rags, packaging materials, and other miscellaneous items. Deactivation waste would include the disposal of process equipment and other materials. All of these materials would be categorized according to existing DOE Orders and ANL-W waste management procedures. The anticipated

categorization of waste types and their expected interim storage and final disposal locations are given in Table 4–9 (see Section 4.2.6). The quantities of various waste forms generated as a result of electrometallurgical treatment at ANL-W are provided in **Table 4–18**.

Table 4–18 Amounts of Waste Generated Under Alternative 1[a]

Waste Stream	*Waste Quantities*	
	Volume (cubic meters)	*Mass (kilograms)*
Direct Process Waste		
Fuel assembly hardware (low-level radioactive waste)	12.5	6,600
High-level radioactive ceramic waste	78 (125 canisters) [b]	120,000
High-level radioactive metallic waste	3.1 (5 canisters) [b]	9,000
Other Associated Process Waste		
Low-level radioactive waste [c]	706	143,000
Transuranic waste	12.5	5,400
Mixed waste	35.3	19,000
Sanitary waste	4,960	1.72×10^6
Deactivation Waste		
Low-level radioactive waste [c]	143	48,000
Transuranic waste	1.6	853
Mixed waste	4.2	2,900

[a] These waste generation estimates are through the year 2015. This is the assumed date that these materials might be sent to the repository. Treatment, high-level radioactive waste processing, deactivation, and interim storage would be accomplished during this time period.
[b] Standardized canisters.
[c] The volumes listed represent final disposal volumes following volume reduction at the Waste Experimental Reduction Facility at INEEL.
Source: ANL 1999.

Estimates of the total amount of other associated process waste that would be generated are based on an evaluation of waste forecasts from ANL-W that account only for the fraction of total ANL-W waste that would be attributable to the processing of sodium-bonded spent nuclear fuel under this alternative. The values in Table 4–18 are for disposal and account for volume reduction. It is anticipated that a large fraction of the low-level radioactive waste generated as a result of electrometallurgical treatment could be volume-reduced at the Waste Experimental Reduction Facility at INEEL prior to disposal at the INEEL Subsurface Disposal Area at the Radioactive Waste Management Complex.

The waste values in Table 4–18 are total quantities that would be generated as a result of Alternative 1 operations. They are not incremental increases over the volumes provided in Table 4–10 that would result from the No Action Alternative. In Alternative 1, the sodium-bonded spent nuclear fuel would be transformed into high-level radioactive waste forms (ceramic and metallic) for disposal in the repository, and in this conversion process, the total volume of material to be disposed of in the repository would be reduced from direct disposal values.

Direct Process Waste

For electrometallurgical treatment, fuel assembly hardware would be removed from the fuel elements in the Fuel Conditioning Facility air cell and disposed of as low-level radioactive waste. These components are primarily stainless steel materials that contain short-lived radionuclides. This waste stream has been

produced at ANL-W for many years and would be handled, as in the past, according to DOE Orders and ANL-W waste management procedures.

Under Alternative 1, metallic and ceramic high-level radioactive waste would be a primary product. The salt removed from the electrorefiners would contain the majority of fission products and transuranics from the spent nuclear fuel. This removed salt would be packaged and transferred to the Hot Fuel Examination Facility for processing into ceramic waste. The metallic waste form would consist primarily of stainless steel cladding hulls containing the noble metal fission products. The hulls would be removed from the electrorefiner and packaged for shipment to the Hot Fuel Examination Facility for processing into the metallic waste form. Both the ceramic and metallic waste would be categorized as high-level radioactive waste. The volumes of waste forms provided in Table 4–18 are for the standardized canisters required for disposal of these materials.

The metallic and ceramic high-level radioactive waste generated would be stored temporarily at the Radioactive Scrap and Waste Facility at ANL-W to allow retrieval for future disposal. The Radioactive Scrap and Waste Facility was designed and constructed for temporary storage of this type of waste. Shielding would be provided by a combination of (1) steel storage liners which would store the waste, and (2) the soil surrounding the liners. When a geologic repository is available, the waste cans containing the metallic and ceramic high-level radioactive waste would be removed from storage, shipped to the INEEL Dry Transfer Facility, and prepared for shipment to the repository.

Other Associated Process Low-Level Radioactive Waste

Low-level radioactive waste would be generated as a result of electrometallurgical treatment of sodium-bonded spent nuclear fuel at ANL-W. This would result from activities in the Fuel Conditioning Facility and the Hot Fuel Examination Facility (e.g., equipment decontamination and repair), as well as in other facilities at ANL-W (e.g., analytical laboratory activities). Material in this waste stream has been generated and routinely handled at ANL-W for many years.

The volume of low-level radioactive waste resulting from electrometallurgical treatment at ANL-W that would require disposal (after volume reduction) would be approximately 48 cubic meters (1,695 cubic feet) per year. This represents approximately 0.08 percent of the total annual volume of low-level radioactive waste currently being disposed of at the INEEL Subsurface Disposal Area at the Radioactive Waste Management Complex, and the total of 706 cubic meters (24,932 cubic feet) represents approximately 0.9 percent of the total Radioactive Waste Management Complex disposal capacity.

Other Associated Process Transuranic Waste

Transuranic waste would be generated by decontamination activities for repair and maintenance of items, and miscellaneous work associated with the electrometallurgical processing. Transuranic waste would be generated primarily from activities conducted in gloveboxes and hot cells at ANL-W.

All of the transuranic waste generated would be packaged and certified in accordance with Waste Isolation Pilot Plant Acceptance Criteria prior to transport to the Waste Isolation Pilot Plant. The transuranic waste generated would amount to approximately 1 cubic meter (35 cubic feet) per year, which is less than 0.002 percent of the volume of transuranic waste in retrievable storage at the Radioactive Waste Management Complex at INEEL. The total volume of transuranic waste is 12.5 cubic meters (441 cubic feet), which is less than 0.008 percent of the estimated total volume of transuranic waste to be placed at the Waste Isolation Pilot Plant.

Other Associated Process Mixed Waste

Mixed waste of this category would be generated primarily from the disposal of any cadmium-contaminated equipment or cleanup material and the analysis of cadmium samples. Mixed waste would be handled according to ANL-W procedures that require limited accumulation at the point of generation. Interim storage of this waste would be accomplished at the Radioactive Scrap and Waste Facility prior to eventual disposal. The Radioactive Scrap and Waste Facility is a permitted mixed waste storage facility for these materials. The mixed waste streams that contribute to the overall mixed waste generated by electrometallurgical treatment have been identified in the INEEL Site Treatment Plan (DOE 1995b).

Deactivation Waste

A variety of waste would be generated as part of deactivation activities associated with electrometallurgical treatment processing at ANL-W. This would include process equipment and process material, such as electrorefiner cadmium. Generated waste categories would include low-level radioactive waste, transuranic waste, and mixed waste. This waste would be categorized and disposed of according to DOE Orders and ANL-W radioactive waste management procedures, as described above for each waste category.

The largest volume of deactivation waste would be low-level radioactive waste, transuranic waste, and mixed waste generated as a result of equipment dismantling and disposal. Components that would require disposition include two electrorefiners, two hot isostatic presses, and two V-mixers, as well as other components such as the grinder/crusher. Decontamination of these components would generate additional mixed, transuranic, and low-level radioactive waste that would require management. The deactivation waste volume would be generated in a single year. This waste would represent an increase of approximately 3.5 times the annual waste generated by electrometallurgical treatment requiring disposal. The total deactivation waste would represent approximately 20 percent over the total associated process waste (excluding sanitary waste) requiring disposal.

4.4 Alternative 2: Clean and Package Blanket Fuel in High-Integrity Cans and Electrometallurgically Treat Driver Fuel at ANL-W

Under this alternative, the sodium-bonded blanket spent nuclear fuel would be cleaned to remove metallic sodium and placed in high-integrity cans. These cans then would be placed into overpack containers prior to dry storage at the Radioactive Scrap and Waste Facility, pending repackaging and transportation for disposal in a geologic repository. The removed sodium contains radioactive elements, principally cesium. The cesium would be separated from the sodium and stabilized as ceramic waste. The sodium would be stabilized using an oxidation/carbonation process (see Appendix C for more detail) (ANL 1999). The sodium-bonded driver spent nuclear fuel would be treated at ANL-W using the electrometallurgical treatment process. The process steps for the electrometallurgical treatment of driver spent nuclear fuel would be similar to those described earlier in Section 4.3 and in Appendix C. The treatment of driver spent nuclear fuel using the electrometallurgical process could start as early as 2000 and could be completed by 2006 to 2007. The preparation of blanket spent nuclear fuel and its placement in high-integrity cans could start in 2003 and could be completed by 2009. In addition, a full year of operations would be needed to deactivate the processing equipment and establish an industrially safe configuration.

4.4.1 Air Quality

Nonradiological Gaseous Emissions

It is expected that criteria and hazardous air pollutants released from operational activities at ANL-W under this alternative would be the same as for Alternative 1, as described in Section 4.3.1 (see also Appendix E, Section E.5.3.1, for more detail).

Radiological Gaseous Emissions

The cleaning of the blanket spent nuclear fuel to remove metallic sodium and the electrometallurgical treatment of the driver spent nuclear fuel would release gaseous fission products to the argon cell environment. Krypton-85 and elemental tritium are the most prevalent radioactive gaseous fission products that would be released to the environment. The released tritium in the cell would not be oxidized due to a very low presence of oxygen and humidity in the argon cell. Appendix E, Section E.4.2, provides details on releases during the processing period at ANL-W. The argon cell also contains an equilibrium concentration of other radionuclide isotopes. Appendix E, Section E.4.1, provides a list of various isotopes that are present in the argon cell in nanocuries (10^{-9} curies) and are released to the atmosphere through the facility stack, along with krypton-85 and elemental tritium. The maximum release of radioactive gaseous emissions would occur when cleaning blanket spent nuclear fuel for placement in high-integrity cans and electrometallurgical treatment of driver spent nuclear fuel are performed simultaneously. This simultaneous operation was estimated to occur over a three-year period starting in 2003. Based on an annual cleaning throughput of 10 metric tons of heavy metal of blanket spent nuclear fuel and an electrometallurgical treatment processing of about 0.6 metric tons of heavy metal of driver spent nuclear fuel, about 809 curies of elemental tritium and 11,860 curies of gaseous krypton-85 would be released annually to the atmosphere. The radiological exposures to the public and workers from radioactive emissions are presented in Section 4.4.4.

4.4.2 Water Resources

Surface Water

No surface water is used at ANL-W. Flood waters from the Big Lost River would not be expected to reach the facilities at ANL-W, as shown in Figure 3–3.

Nonradiological Liquid Effluent

There are no discharges to the surface waters at ANL-W, except for discharges of nonhazardous liquid waste to the sewage pond and to the industrial waste pond. Big Lost River, Little Lost River, and Birch Creek would not be impacted by activities associated with high-integrity can operations and electrometallurgical treatment process operations. Current operating and monitoring practices would continue for NPDES stormwater and liquid effluent discharges associated with facilities at ANL-W (see also Section 4.2.2).

During fuel treatment and associated activities, some hazardous materials may be used inside buildings. To prevent potential releases to surface or subsurface waters resulting from spills of hazardous materials used in buildings, facilities are designed, constructed, and maintained to contain these materials. Double-contained pipes, leak detection, and secondary containment of tanks are some of the features used to prevent hazardous material releases to the environment. Following existing written procedures, spill containment and cleanup equipment is present in areas where hazardous materials are stored or used (DOE 1996b).

Radiological Liquid Effluent

No radiological liquid effluent or waste generated by high-integrity can and electrometallurgical treatment process operations would be discharged to surface water.

Groundwater

Under this alternative at ANL-W, there would be little change in groundwater consumption for domestic use since there is little change expected in the number of workers. The current water usage at ANL-W is 188 million liters (49.6 million gallons) per year.

Nonradiological Liquid Effluent

No nonradiological liquid effluent generated by high-integrity can and electrometallurgical treatment process operations would be discharged to groundwater.

Radiological Liquid Effluent

No radiological liquid effluent or waste generated by high-integrity can and electrometallurgical treatment process operations would be discharged to groundwater.

4.4.3 Socioeconomics

Under this alternative, the existing facilities at ANL-W would remain operational. No new employment or in-migration of workers would be required. Thus, there would be no additional impacts on the socioeconomic conditions in the region around ANL-W and INEEL.

4.4.4 Public and Occupational Health and Safety

The assessments of potential incremental radiological and chemical impacts associated with this alternative are presented in this section. Summaries of radiological impacts from normal operations are presented in Tables 4–19 and 4–20 for the public and workers, respectively. The radiological impacts from a spectrum of hypothetical accident scenarios are provided in Tables 4–21 and 4–22. The impacts from hazardous chemical releases during accident conditions are presented in Table 4–23. Background information on the effects of radiation on human health and safety is presented in Appendix E, Section E.2.

4.4.4.1 Normal Operations

Radiological Impacts

Under this alternative, radioactive releases would occur during sodium-bonded blanket spent nuclear fuel cleaning and driver spent nuclear fuel chopping. All of these activities would be performed in the argon cell. Appendix E, Sections E.3, E.4.1, and E.4.2, details the method and assumptions used for calculating the impacts of normal operational radiological releases on the public health and safety. The maximum annual dose to the public would result when cleaning of blanket spent nuclear fuel and treatment of driver spent nuclear fuel are performed simultaneously under this alternative. Appendix E, Section E.4.2, provides details on the treatment process duration and throughputs for each fuel type. The duration of the treatment process is estimated to be nine years.

Calculated maximum annual and project total radiological impacts to the public are given in **Table 4–19**. The impacts are calculated for two types of receptors: the general public living within 80 kilometers (50 miles) of ANL-W in 2010, and a maximally exposed offsite individual (a member of the public assumed to be residing at the INEEL site boundary and receiving the maximum dose). Primary contributors to doses to members of the public are releases of tritium gas (about 1 percent of which was conservatively assumed to be in oxidized form) and krypton-85, which together contribute over 99.9 percent of the total calculated doses. To put the operational impacts into perspective, comparisons with impacts from natural background radiation also are included in the table. As shown in this table, the expected radiation doses to the maximally exposed offsite individual and the general public are much smaller than the limit of 10 millirem per year set by the EPA (40 CFR 61) and DOE (DOE Order 5400.5).

Table 4–19 Annual and Project Total Radiological Impacts to the Public From Operational Activities Under Alternative 2

Receptor	*Electrometallurgically Treat Driver Spent Nuclear Fuel*	*Clean and Place Blanket Spent Nuclear Fuel in High-Integrity Cans*	*Total*
Population Within 80 Kilometers (50 Miles) in the Year 2010			
Collective dose (person-rem per year) [a]	0.0027	0.00028	0.0030
Excess latent cancer fatalities (per year)	1.4×10^{-6}	1.4×10^{-7}	1.5×10^{-6}
Project total excess latent cancer fatalities [b]	8.0×10^{-6}	3.4×10^{-7}	8.3×10^{-6}
Maximally Exposed Offsite Individual			
Dose (millirem per year) [a]	0.00033	0.000048	0.00038
Percent of annual background [c]	0.000092	0.000013	0.00011
Latent cancer fatality risk (per year)	1.7×10^{-10}	2.4×10^{-11}	1.9×10^{-10}
Project total lifetime cancer fatality risk	9.4×10^{-10}	5.8×10^{-11}	1.0×10^{-9}
Average Individual Within 80 Kilometers (50 Miles)			
Dose (millirem per year) [d]	0.000011	1.2×10^{-6}	0.000012
Latent cancer fatality risk (per year)	5.6×10^{-12}	5.8×10^{-13}	6.2×10^{-12}
Project total lifetime cancer fatality risk [b]	3.3×10^{-11}	1.4×10^{-12}	3.5×10^{-11}

[a] Annual maximum dose during normal operations.
[b] Total calculated risk over nine years.
[c] The annual natural background radiation level at INEEL is about 360 millirem for the average individual (see Table 3–8); the population within 80 kilometers (50 miles) in the year 2010 would receive 86,500 person-rem.
[d] Obtained by dividing the population dose by the number of people projected to live within 80 kilometers (50 miles) of ANL-W in the year 2010 (240,338).

Table 4–20 summarizes worker population doses. Occupational doses were estimated by examining the type and duration of various operations performed by ANL-W workers involved with sodium-bonded spent nuclear fuel high-integrity can and electrometallurgical treatment processes. It was concluded that the average worker dose would not be different from that currently being experienced. The estimated annual collective worker dose would be 22 person-rem, with an average individual dose of 60 millirem per year for each of the 346 involved workers. If these estimates were extended over the nine years of treatment activities, plus one year for deactivation of the facilities, the project total worker population dose would be 231 person-rem, leading to a risk of 0.092 latent cancer fatalities.

As shown in Tables 4–19 and 4–20:

- The annual dose to the maximally exposed offsite individual would be 0.00038 millirem per year, with an associated 1.9×10^{-10} risk per year of developing a fatal cancer (or one chance in 5.3 billion that the individual would develop a fatal cancer per year of exposure).

- The collective dose to the population within 80 kilometers (50 miles) of the ANL-W facilities would be 0.0030 person-rem per year, with an associated 1.5×10^{-6} latent cancer fatalities per year (or one chance in 666,700 that the population would experience a fatal cancer per year of exposure).

- The project total risk to the population within 80 kilometers (50 miles) of the ANL-W facilities would be 8.3×10^{-6} latent cancer fatalities (or one chance in 120,000 that the exposed population would experience a fatal cancer).

Table 4–20 Annual and Project Total Radiological Impacts to Workers From Operational Activities Under Alternative 2

Receptor	*Impacts*
Worker [a]	
Average worker dose (millirem per year)	60
Average worker latent cancer fatality risk (project total over nine years)	0.00022
Worker Population	
Collective dose (person-rem per year) [b]	22
Excess latent cancer fatalities (per year)	0.0088
Project total dose (person-rem)	231
Project total excess latent cancer fatalities	0.092

[a] The regulatory dose limit for an individual worker is 5,000 millirem per year (10 CFR 835). However, the maximum annual dose to a worker would be kept below the DOE Administrative Control Level of 2,000 millirem per year, as established for all DOE activities in DOE Order N 441.1.

[b] Worker dose is 33 person-rem for one year of deactivation activities.

Source: ANL 1999.

- The collective dose to facility workers would be 22 person-rem per year, with an associated 0.0088 latent cancer fatalities per year (or one chance in 113 that the workers would experience a fatal cancer per year of operation).

- The project total dose to facility workers would be 231 person rem, with an associated 0.092 latent cancer fatalities (or one chance in 11 that the exposed workers would experience a fatal cancer).

These results indicate that no increase in the expected number of fatal cancers among the various affected populations would result from the activities under this alternative.

Hazardous Chemical Impacts

The hazardous chemical impacts to the public residing at the INEEL site boundary from the operational activities at ANL-W under Alternative 2 would be similar to the impacts evaluated for Alternative 1 described in Section 4.3.4.1. The results indicate that no adverse toxic health or cancer effects would be expected from exposure to hazardous chemicals released under this alternative. The existing chemical environment is described in Section 3.2.10.2.

4.4.4.2 Facility Accidents

Radiological Impacts

Potential radiological impacts to the public and a noninvolved onsite worker due to accidents during cleaning activities for placement of blanket spent nuclear fuel elements in high-integrity cans and the electrometallurgical treatment operational activities for driver spent nuclear fuel are summarized and presented in this section. The detailed analysis of facility accidents, with the associated assumptions, is presented in Appendix F.

The detailed analysis considered a wide spectrum of potential accident scenarios including fire, spills, criticality, earthquake, and aircraft crash. The aircraft crash and criticality events were determined to have an occurrence frequency of less than 10^{-7} per year, and consequence analyses for these two events were not

performed. Cleaning of the blanket spent nuclear fuel would be performed in the Hot Fuel Examination Facility; treatment of the driver spent nuclear fuel would be performed in both the Hot Fuel Examination Facility and the Fuel Conditioning Facility. Because driver spent nuclear fuel processing would take place in both of these facilities, the beyond-design-basis earthquake event is assessed for the driver spent nuclear fuel, taking into account the multifacility impacts of this event. The cleaning of the blanket spent nuclear fuel would be performed only in the Hot Fuel Examination Facility. **Table 4–21** presents the frequencies and consequences of the postulated set of accidents to the maximally exposed offsite individual, the offsite population residing within 80 kilometers (50 miles) of the facility, and a noninvolved worker located 100 meters (330 feet) to 230 meters (755 feet) from the facility.

Table 4–21 Accident Frequency and Consequences at ANL-W Under Alternative 2

Accident	*Frequency (event per year)*	*Maximally Exposed Offsite Individual*		*Population Within 80 Kilometers (50 Miles)*		*Noninvolved Worker*	
		Dose (millirem)	*Latent Cancer Fatality Risk* [a]	*Dose (person-rem)*	*Excess Latent Cancer Fatalities* [b]	*Dose (millirem)*	*Latent Cancer Fatality Risk* [a]
Driver Spent Nuclear Fuel							
Salt powder spill	0.01	0.00046	2.3×10^{-10}	0.000098	4.9×10^{-8}	4.7×10^{-7}	1.9×10^{-13}
Salt transfer drop	1.0×10^{-7}	0.19	9.5×10^{-8}	0.022	0.000011	0.073	2.9×10^{-8}
Transuranic waste fire	0.001	0.059	3.0×10^{-8}	0.0071	3.6×10^{-6}	0.22	8.8×10^{-8}
Cask drop	0.01	0.030	1.5×10^{-8}	0.0035	1.8×10^{-6}	0.00084	3.4×10^{-10}
Design-basis earthquake	0.008	12	6.0×10^{-6}	1.4	0.0007	4.7	1.9×10^{-6}
Beyond-design-basis earthquake	0.00001	22,000	0.022	2,500	1.3	370	0.00015
Blanket Spent Nuclear Fuel							
Cask drop	0.01	0.0024	1.2×10^{-9}	0.00028	1.4×10^{-7}	0.000049	2.0×10^{-11}
Transuranic waste fire	0.001	0.059	3.0×10^{-8}	0.0071	3.6×10^{-6}	0.22	8.8×10^{-8}
Sodium fire [c]	0.008	5.9	3.0×10^{-6}	0.69	0.00035	0.054	2.2×10^{-8}

[a] Increased likelihood of a latent cancer fatality.
[b] Increased number of latent cancer fatalities.
[c] The frequency for this accident is the frequency for the facility design-basis earthquake initiating cell fire.

The dose to the maximally exposed offsite individual was calculated for the 95th percentile meteorological conditions. The doses to the population and the noninvolved worker were calculated using 50th percentile meteorological conditions. DOE did not quantitatively estimate the involved worker dose due to accidents (see the discussion on the involved worker in Section 4.2.4.2). The accident risks for the same receptors are summarized in **Table 4–22**.

For accidents at ANL-W, the highest risk of a latent cancer fatality to the maximally exposed offsite individual and to a noninvolved worker would be 2.2×10^{-7} per year (or one chance in 4.5 million that the individual would develop a fatal cancer per year of operation) and 1.5×10^{-8} per year (or one chance in 66.7 million that the individual would develop a fatal cancer per year of operation), respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.000013 per year (or one chance in 76,920 that the population would develop a fatal cancer per year of operation).

Table 4–22 Annual Cancer Risks Due to Accidents at ANL-W Under Alternative 2

Accident	*Maximally Exposed Offsite Individual* [a]	*Population Within 80 Kilometers (50 Miles)* [b]	*Noninvolved Worker* [a]
Driver Spent Nuclear Fuel			
Salt powder spill	2.3×10^{-12}	4.9×10^{-10}	1.9×10^{-15}
Salt transfer drop	9.5×10^{-15}	1.1×10^{-12}	2.9×10^{-15}
Transuranic waste fire	3.0×10^{-11}	3.6×10^{-9}	8.8×10^{-11}
Cask drop	1.5×10^{-10}	1.7×10^{-8}	3.4×10^{-12}
Design-basis earthquake	4.8×10^{-8}	5.6×10^{-6}	1.5×10^{-8}
Beyond-design-basis earthquake	2.2×10^{-7}	0.000013	1.5×10^{-9}
Blanket Spent Nuclear Fuel			
Cask drop	1.2×10^{-11}	1.4×10^{-9}	2.0×10^{-13}
Transuranic waste fire	3.0×10^{-11}	3.6×10^{-9}	8.8×10^{-11}
Sodium fire	2.4×10^{-8}	2.8×10^{-6}	1.7×10^{-10}

[a] Increased likelihood of a latent cancer fatality.
[b] Increased number of latent cancer fatalities.

Hazardous Chemical Impacts

Nonradiological impacts are evaluated in terms of comparison to ERPG. ERPG values are estimates of airborne concentration thresholds above which one can reasonably anticipate observing adverse effects (see Appendix F, Section F.3.1.2, for details).

The hazardous chemical impacts of potential facility accidents associated with the treatment of driver spent nuclear fuel using the electrometallurgical treatment process are summarized in **Table 4–23**.

Table 4–23 Hazardous Chemical Impacts Due to Accidents at ANL-W Under Alternative 2

Accident	*Frequency (event per year)*	*Receptor*	*Exposure*
Uranium handling accident	0.01	Noninvolved worker	Uranium: less than ERPG-1
		Maximally exposed offsite individual	Uranium: less than ERPG-1
Design-basis earthquake	0.0002	Noninvolved worker	Uranium: less than ERPG-1
		Maximally exposed offsite individual	Uranium: less than ERPG-1
Sodium fire	0.008	Noninvolved worker	Sodium: less than ERPG-1
		Maximally exposed offsite individual	Sodium: less than ERPG-1
Uranium fire	0.00001	Noninvolved worker	Uranium: less than ERPG-1
		Maximally exposed offsite individual	Uranium: less than ERPG-1
Beyond-design-basis earthquake	0.00001	Noninvolved worker	Cadmium: less than ERPG-1
			Uranium: less than ERPG-1
		Maximally exposed offsite individual	Cadmium: less than ERPG-1
			Uranium: less than ERPG-1

ERPG = Emergency Response Planning Guideline.

4.4.5 Environmental Justice

As discussed in Appendix H, Executive Order 12898 directs Federal agencies to address disproportionately high and adverse health or environmental effects of alternatives on minority or low-income populations.

Analyses of normal operations and accident conditions presented in Section 4.4.4 show the risk of latent cancer fatalities to the public residing within 80 kilometers (50 miles) of the electrometallurgical treatment processing facilities at ANL-W to be much lower than 1. Therefore, radiological and nonradiological risks posed by implementation of this alternative would have no disproportionately high and adverse consequences on any particular group within the general population, including minority or low-income populations.

4.4.6 Waste Management

This alternative would generate process waste from treatment operations, other associated process waste from normal support operations, and deactivation waste following the conclusion of operations. Process waste would include fuel assembly hardware and high-level radioactive metallic and ceramic waste. Other associated process waste would include operational waste such as failed equipment, rags, packaging materials, and other miscellaneous items. Deactivation waste would include the disposal of process equipment and other materials. All of these materials would be categorized according to existing DOE Orders and ANL-W waste management procedures. The anticipated categorization of these waste types and their expected interim storage and final disposal locations are given in Table 4–9 (see Section 4.2.6). The quantities of various waste forms generated as a result of Alternative 2 are provided in **Table 4–24**.

Table 4–24 Amounts of Waste Generated Under Alternative 2[a]

Waste Stream	*Waste Quantities*	
	Volume (cubic meters)	*Mass (kilograms)*
Direct Process Waste		
Fuel assembly hardware (low-level radioactive waste)	12.5	6,000
High-level radioactive ceramic waste	16.3 (26 canisters) [b]	24,400
High-level radioactive metallic waste	1.3 (2 canisters) [b]	2,500
Spent nuclear fuel	25.2 (63 canisters) [b]	63,000
Other Associated Process Waste		
High-level radioactive waste	0.4 (1 canister) [b]	220
Low-level radioactive waste [c]	555	113,000
Transuranic waste	9.1	3,800
Mixed waste	27.5	14,800
Sanitary waste	4,960	1.72×10^6
Deactivation Waste		
Low-level radioactive waste [c]	166.2	56,000
Transuranic waste	1.6	853
Mixed waste	4.8	3,200

[a] These waste generation estimates are through the year 2015. This is the assumed date that these materials might be sent to the repository. Treatment, high-level radioactive waste processing, deactivation, and interim storage would be accomplished during this time period.

[b] Standardized canisters.

[c] The volumes listed represent final disposal volumes following volume reduction at the Waste Experimental Reduction Facility at INEEL.

Source: ANL 1999.

Estimates of the total amount of other associated process waste that would be generated are based on an evaluation of waste forecasts from ANL-W that accounted only for the fraction of total ANL-W waste that would be attributable to the treatment of sodium-bonded spent nuclear fuel. The values in Table 4–24 are for disposal and account for volume reduction. It is anticipated that a large fraction of the low-level radioactive waste that would be generated as a result of Alternative 2 could be volume-reduced at the Waste Experimental Reduction Facility at INEEL prior to disposal at the INEEL Subsurface Disposal Area at the Radioactive Waste Management Complex.

The waste values in Table 4–24 are total quantities that would be generated as a result of Alternative 2 operations. They are not incremental increases over the volumes provided in Table 4–10 that would result from the No Action Alternative. In Alternative 2, the driver spent nuclear fuel would be transformed into high-level radioactive waste forms (ceramic and metallic) for disposal in the repository. The blanket spent nuclear fuel would be cleaned and packaged in high-integrity cans for disposal in the repository. In this conversion process, the total volume of material to be disposed of in the repository would be reduced from direct disposal values.

Direct Process Waste

For this alternative, fuel hardware would be removed from the fuel elements in the Fuel Conditioning Facility air cell and disposed of as low-level radioactive waste. These components are primarily stainless steel materials that contain short-lived radionuclides. This waste stream has been produced at ANL-W for many years and would be handled, as in the past, according to DOE Orders and ANL-W waste management procedures.

Under this alternative, metallic and ceramic high-level radioactive waste would be a primary product of the electrometallurgical treatment of driver spent nuclear fuel. The salt removed from electrorefiners would contain the majority of fission products and transuranics from the spent nuclear fuel. This removed salt would be packaged and transferred to the Hot Fuel Examination Facility for processing into ceramic waste. The metallic waste form would consist primarily of stainless steel cladding hulls containing the noble metal fission products. The hulls would be removed from the electrorefiner and packaged for shipment to the Hot Fuel Examination Facility for processing into the metallic waste form. Both the ceramic and metallic waste would be categorized as high-level radioactive waste.

The packaged spent nuclear fuel volume is based on placing the blanket spent nuclear fuel in high-integrity cans which would be placed in standardized canisters. The volumes of waste forms provided in Table 4–24 are for the standardized canisters required for disposal of these materials.

The metallic and ceramic high-level radioactive waste generated as a result of electrometallurgical treatment of driver spent nuclear fuel under this alternative, driver and blanket spent nuclear fuel generated during the demonstration project, and packaged spent nuclear fuel would be stored temporarily at the Radioactive Scrap and Waste Facility at ANL-W to allow retrieval for future disposal. The Radioactive Scrap and Waste Facility was designed and constructed for temporary storage of this type of waste. Shielding would be provided by a combination of (1) steel storage liners which would store the waste, and (2) the soil surrounding the liners. When a geologic repository is available, the waste cans containing these materials would be removed from storage, shipped to the INEEL Dry Transfer Facility, and prepared for shipment to the repository.

Other Associated Process High-Level Radioactive Waste

High-level radioactive waste could be generated as a result of blanket spent nuclear fuel processing at ANL-W. This would result from activities in the Hot Fuel Examination Facility. Material in this waste

stream would consist of the absorbent used in the off-gas system which has collected the volatile radionuclides released from the spent nuclear fuel when heated.

The volume of high-level radioactive waste generated is expected to be less than the amount needed to fill a single standardized waste canister. Conservatively, the volume of a single canister, 0.4 cubic meters, has been used for the volume of high-level radioactive waste generated.

Other Associated Process Low-Level Radioactive Waste

Low-level radioactive waste would be generated as a result of processing at ANL-W. This would result from activities in the Fuel Conditioning Facility and the Hot Fuel Examination Facility (e.g., equipment decontamination and repair), as well as in other facilities at ANL-W (e.g., analytical laboratory activities). Material in this waste stream has been generated and routinely handled at ANL-W for many years.

The volume of low-level radioactive waste at ANL-W that would require disposal (after volume reduction) would be approximately 50 cubic meters (1,766 cubic feet) per year. This represents approximately 1 percent of the total annual volume of low-level radioactive waste currently being disposed of at the INEEL Subsurface Disposal Area in the Radioactive Waste Management Complex, and the total of 555 cubic meters (19,600 cubic feet) represents approximately 0.5 percent of the total Radioactive Waste Management Complex disposal capacity.

Other Associated Process Transuranic Waste

Transuranic waste would be generated by Alternative 2 from decontamination activities for repair and maintenance of items, and miscellaneous work associated with processing the sodium-bonded spent nuclear fuel. Transuranic waste would be generated primarily from activities conducted in gloveboxes and hot cells at ANL-W.

All of the transuranic waste generated as a result of the treatment of sodium-bonded spent nuclear fuel at ANL-W would be packaged and certified in accordance with Waste Isolation Pilot Plant Acceptance Criteria prior to transport to the Waste Isolation Pilot Plant. The transuranic waste generated would amount to approximately 1 cubic meter (35 cubic feet) per year, which is less than 0.002 percent of the volume of transuranic waste in retrievable storage at the Radioactive Waste Management Complex at INEEL. The total volume of transuranic waste is 9.1 cubic meters (321 cubic feet), which is 0.005 percent of the estimated total volume of transuranic waste to be placed at the Waste Isolation Pilot Plant.

Other Associated Process Mixed Waste

Mixed waste would be generated primarily from the disposal of any cadmium-contaminated equipment or cleanup material and the analysis of cadmium samples. Mixed waste would be handled according to ANL-W procedures that require limited accumulation at the point of generation. Interim storage of this waste would be accomplished at the Radioactive Scrap and Waste Facility prior to eventual disposal. The Radioactive Scrap and Waste Facility is a permitted mixed waste storage facility for these materials. The mixed waste streams that contribute to the overall mixed waste generated by electrometallurgical treatment have been identified in the INEEL Site Treatment Plan (DOE 1995b).

Deactivation Waste

A variety of waste would be generated as part of deactivation activities associated with electrometallurgical treatment processing at ANL-W. This would include process equipment and process material, such as electrorefiner cadmium from electrometallurgical treatment of driver spent nuclear fuel. Generated waste

categories would include low-level radioactive waste, transuranic waste, and mixed waste. This waste would be categorized and disposed of according to DOE Orders and ANL-W radioactive waste management procedures, as described above for each waste category.

The largest volume of deactivation waste would be low-level radioactive waste generated as a result of equipment dismantling and disposal. Components of electrometallurgical treatment of the driver spent nuclear fuel that would require disposition include two electrorefiners, two hot isostatic presses, and two V-mixers, as well as other components such as the grinder/crusher. Decontamination of these components would generate additional mixed, transuranic, and low-level radioactive waste that would require management. The deactivation waste volume would be generated in two years. The total deactivation waste would represent an additional 30 percent over the total associated process waste (excluding sanitary waste) requiring disposal.

4.5 ALTERNATIVE 3: DECLAD AND CLEAN BLANKET FUEL AND ELECTROMETALLURGICALLY TREAT DRIVER FUEL AT ANL-W; PUREX PROCESS BLANKET FUEL AT SRS

Under this alternative, the sodium-bonded blanket spent nuclear fuel would be declad and cleaned to remove metallic sodium, packaged in aluminum cans at ANL-W, and shipped to SRS for treatment using the PUREX process at F-Canyon. The removed sodium would be stabilized using an oxidation/carbonation process (ANL 1999). The sodium-bonded driver spent nuclear fuel would be treated at ANL-W using the electrometallurgical treatment process. The high-level radioactive waste generated from the treatment of the blanket spent nuclear fuel at SRS would be in the form of borosilicate glass and would be stored at the SRS Defense Waste Processing Facility, pending repackaging and transportation for disposal in a geologic repository. The process steps for the electrometallurgical treatment of driver spent nuclear fuel would be similar to those described earlier in Section 4.3 and in Appendix C. The treatment of driver spent nuclear fuel using the electrometallurgical process could start as early as 2000 and could be completed by 2006 to 2007. The preparation of blanket spent nuclear fuel and its shipment to SRS could start in 2003 and could be completed by 2009. In addition, a full year of operations would be needed to deactivate the processing equipment and establish an industrially safe configuration.

PUREX processing of blanket spent nuclear fuel at SRS would require six months of operation and could be completed by 2010.

4.5.1 Air Quality

Nonradiological Gaseous Emissions

It is expected that criteria and hazardous air pollutants released from operational activities at ANL-W under this alternative to be the same as for Alternative 1, as described in Section 4.3.1, (see also Appendix E, Section E.5.3.1 for more detail). Baseline air quality concentrations are presented in Section 3.2.3.1.

The concentrations of nonradiological air pollutants attributed to this alternative at SRS are presented in **Table 4–25**. These concentrations are based on information in the *Savannah River Site Spent Nuclear Fuel Management Final Environmental Impact Statement* (SRS Spent Nuclear Fuel Management Final EIS) (DOE 2000) for the PUREX processing of similar fuel. See Appendix E, Section E.5.3.2, for more details. The site boundary concentrations are equal to the incremental concentrations generated in this alternative plus the baseline concentrations given in Section 3.3.3.1. Only those air pollutants that are expected and have ambient air quality standards are presented in the table. Note that there are no Prevention of Significant Deterioration increment-consuming sources at SRS; therefore, a Prevention of Significant Deterioration increment analysis was not performed.

Table 4–25 Nonradiological Air Quality Concentrations at the Site Boundary Under Alternative 3 at SRS for Comparison With Ambient Air Quality Standards

Pollutant	*Averaging Period*	*Most Stringent Standard or Guideline (micrograms per cubic meter)*	*Maximum Incremental Concentration (micrograms per cubic meter)*
Criteria Pollutants			
Carbon monoxide	8 hours 1 hour	10,000 40,000	1.22 9.06
Nitrogen dioxide	Annual	100	3.11
PM_{10}	Annual 24 hours (interim) 24 hours (99^{th} percentile over 3 years)	50 150 150	Less than 0.01 0.11 Not available
$PM_{2.5}$	3-year annual 24 hours (98^{th} percentile over 3 years)	15 65	Not available Not available
Sulfur dioxide	Annual 24 hours 3 hours	80 365 1,300	Less than 0.01 0.12 0.91
State-regulated Pollutants			
Gaseous fluoride	30 days 7 days 24 hours 12 hours	0.8 1.6 2.9 3.7	0.01 0.03 0.06 0.11
Total suspended particulates	Annual	75	Less than 0.01
Hazardous/Toxic Compounds			
1,1,1-trichloroethane	24 hours	9,550	Less than 0.01
Benzene	24 hours	150	0.01
Ethanolamine	24 hours	200	Less than 0.01
Ethyl benzene	24 hours	4,350	Less than 0.01
Ethylene glycol	24 hours	650	Less than 0.01
Formaldehyde	24 hours	15	Less than 0.01
Glycol ethers	24 hours	No standard	Less than 0.01
Hexachloronaphthalene	24 hours	1	Less than 0.01
Hexane	24 hours	900	0.01
Manganese	24 hours	25	Less than 0.01
Methyl alcohol	24 hours	1,310	Less than 0.01
Methyl-ethyl-ketone	24 hours	14,750	Less than 0.01
Methyl-isobutyl-ketone	24 hours	2,050	Less than 0.01
Methylene chloride	24 hours	8,750	0.01
Naphthalene	24 hours	1,250	Less than 0.01
Nitric acid	24 hours	125	0.28
Phenol	24 hours	190	Less than 0.01
Phosphorous	24 hours	0.5	Less than 0.01
Sodium hydroxide	24 hours	50	Less than 0.01
Toluene	24 hours	2,000	0.01
Trichloroethane	24 hours	6,750	Less than 0.01
Vinyl acetate	24 hours	176	Less than 0.01
Xylene	24 hours	4,350	0.02

PM_n = Particulate matter less than or equal to *n* microns in diameter.
Source: Bickford et al. 1997.

Radiological Gaseous Emissions

The decladding and cleaning of blanket spent nuclear fuel and the electrometallurgical treatment of driver spent nuclear fuel at ANL-W would release gaseous fission products to the hot (argon) cell environment. Krypton-85 and elemental tritium are the most prevalent radioactive gaseous fission products that would be released to the environment. The released tritium in the cell would not be oxidized due to a very low presence of oxygen and humidity in the argon cell. The argon cell also contains an equilibrium concentration of other radionuclide isotopes. Appendix E, Section E.4.1, provides a list of various isotopes that are present in the argon cell in nanocuries (10^{-9} curies) and are released to the atmosphere through the facility stack, along with the krypton-85 and elemental tritium. The maximum release of radioactive gaseous emissions would occur when decladding blanket spent nuclear fuel for packaging and shipment to SRS and electrometallurgical treatment of driver spent nuclear fuel are performed simultaneously. This simultaneous operation was estimated to occur over a three-year period starting in 2003. Appendix E, Section E.4.2, provides details on releases during the processing period at ANL-W. Based on an annual decladding throughput of 10 metric tons of heavy metal of blanket spent nuclear fuel elements and an electrometallurgical treatment processing of about 0.6 metric tons of heavy metal of driver spent nuclear fuel elements, about 809 curies of elemental tritium and 11,860 curies of gaseous krypton-85 would be released annually to the atmosphere (see Appendix E, Section E.4.2).

Since declad and clean fuel would be packaged and sent to SRS, some gaseous fission products would be expected to be present in that fuel. However, it was assumed conservatively that all gaseous fission products in the blanket spent nuclear fuel would be released to the environment during PUREX processing at SRS over a six-month period (see Appendix E, Section E.4.3). The radiological exposures to the public and workers from radioactive emissions are presented in Section 4.5.4.

4.5.2 Water Resources

As stated in Section 4.4.2, decladding and cleaning of blanket spent nuclear fuel and treatment of driver spent nuclear fuel using electrometallurgical treatment would not discharge any radiological chemical material to the surface or groundwater at the INEEL site. These activities also would not impact the current groundwater usage at the site. For a discussion of impacts on water resources at ANL-W, see Section 4.4.2.

The impacts on water resources from processing blanket spent nuclear fuel at F-Canyon are described below.

Surface Water

No surface water would be used for PUREX processing of blanket spent nuclear fuel at the F-Area. The F-Canyon processing facilities are outside the 100-year floodplain, as shown in Figure 3-6.

Nonradiological Liquid Effluent

The major sources of liquid effluent from PUREX processing of blanket spent nuclear fuel at SRS would be process cooling water and steam condensate. There are sufficient capacities in existing wastewater treatment facilities to handle the liquid effluent from this processing. Liquid effluent associated with PUREX processes would use these facilities and the existing permitted outfalls (Section 3.3.4.1). Process cooling water treatment would result in releases to Upper Three Runs Creek from the F-Area, as shown in **Table 4–26**. Sanitary waste would be treated at the SRS Central Wastewater Treatment Facility and discharged through an existing NPDES-permitted outfall. Since employment would not increase as a result of processing this fuel, the treatment rates through the Central Wastewater Treatment Facility would not be affected and the requirements of the SRS NPDES permit would continue to be met (DOE 2000).

Table 4–26 Chemical Effluent Concentrations From PUREX Cooling Water Treatment

Parameter	Effluent Concentrations	Existing Stream Water Concentrations		Water Quality Criterion (milligrams per liter) [c]
	F-Area (milligrams per liter)	Upper Three Runs (Upstream) [a] (Average) (milligrams per liter)	Upper Three Runs (Downstream) [b] (Average) (milligrams per liter)	
Aluminum	0.2	0.19	0.24	(d)
Ammonia	0.03	0.0001	Not reported	(d)
Chromium	0.02	Not detected	Not detected	0.1
Copper	0.01	0.018	0.015	1
Manganese	0.01	0.039	0.052	0.05
Nickel	0.05	Not detected	Not detected	0.1
Nitrate	0.04	0.36	0.27	10
Zinc	0.07	0.06	0.091	3

[a] Stream monitor U3R-1A.
[b] Stream monitor U3R-4.
[c] Federal Drinking Water Standards and Health Advisories (EPA 1996) and South Carolina Water Quality Criteria for Protection of Human Health (SCDHEC 1998).
[d] No drinking water standard.
Sources: Arnett and Mamatey 1998, DOE 2000.

Although proposed or final Federal drinking water standards do not apply to the discharges, these standards are used for comparison to SRS discharges. The discharge concentration would not exceed the Federal drinking water standard. The discharges would also comply with the South Carolina Water Classifications and Standards (SCDHEC 1998). The release concentrations would be no greater than the concentrations measured in Upper Three Runs (Arnett and Mamatey 1998), with the exception of zinc and ammonia. Zinc concentrations in the discharge are within the Federal health advisory limits (EPA 1996).

Radiological Liquid Effluent

PUREX processing would release measurable radioactive nuclides to the surface water through the cooling water system. The expected radiological effluent from processing declad and cleaned blanket spent nuclear fuel at F-Canyon was estimated based on the measured data from various effluent streams at F-Area as presented in the SRS Environmental Data for 1997 (Arnett and Mamatey 1998). Since the mechanism associated with releases of liquid effluent from PUREX processing at F-Canyon is essentially the same for almost every fuel type processed, the F-Area 1997 effluent data were used to conservatively represent the potential releases from a six-month operation of F-Canyon. **Table 4–27** provides a list of potential radiological isotopes that could be released to the surface water during processing of approximately 57 metric tons of heavy metal of blanket spent nuclear fuel (see Appendix E, Section E.4.3, for details).

Groundwater

All process water would come from groundwater, as would sanitary water. At most, less than 65 million liters (17 million gallons) per year would be required for cooling water. SRS annually withdraws more than 5 billion liters (more than 1.3 billion gallons) per year of groundwater (DOE 2000).

Table 4–27 Estimated Radiological Liquid Effluent From PUREX Processing of Blanket Spent Nuclear Fuel

Isotope	*Curies*
Tritium (Hydrogen-3)	1.54
Strontium-89/Strontium-90	0.000031
Cesium-137	0.0022
Uranium-234	0.000085
Promethium-147	0.000011
Uranium-238	0.00019
Plutonium-238	0.000016
Plutonium-239	7.8×10^{-6}

Source: Arnett and Mamatey 1998.

Nonradiological Liquid Effluent

No nonradiological chemicals would be discharged to groundwater from PUREX processing of blanket spent nuclear fuel at F-Canyon and the FB-Line in F-Area.

Radiological Liquid Effluent

No radiological liquid effluent or waste would be discharged to groundwater from PUREX processing of blanket spent nuclear fuel at F-Canyon and the FB-Line in F-Area.

4.5.3 Socioeconomics

Under this alternative, the existing facilities at ANL-W and SRS would remain operational. No new employment or in-migration of workers would be required. Thus, there would be no additional impacts on the socioeconomic conditions in the regions around INEEL and SRS.

4.5.4 Public and Occupational Health and Safety

The assessments of potential incremental radiological and chemical impacts associated with this alternative are presented in this section. Summaries of radiological impacts from normal operations are presented in Tables 4–28 and 4–29 for the public and workers, respectively. The radiological impacts from a spectrum of hypothetical accident scenarios are provided in Tables 4–31 through 4–34. The impacts from hazardous chemical releases during accident conditions are presented in Tables 4–35 and 4–36. Background information on the effects of radiation on human health and safety is presented in Appendix E, Section E.2.

4.5.4.1 Normal Operations

Radiological Impacts

Under this alternative, radioactive releases would occur during PUREX processing at F-Canyon. Appendix E, Sections E.3 and E.4.3, details the method and assumptions used for calculating the impacts of normal operational radiological releases on the public health and safety. Doses to the public would result from treating about 57 metric tons of heavy metal of blanket spent nuclear fuel. The blanket spent nuclear fuel being processed at SRS is already declad and cleaned at ANL-W; therefore, the gaseous fission products are assumed to have already been released. However, for the analytical purposes of this EIS, it was

conservatively assumed that the gaseous fission products would still be within the matrix of the fuel and would be released during PUREX processing at SRS. The processing was assumed to continue for six months (see Appendix E.4.3).

Calculated incremental maximum annual and project total radiological impacts to the public are given in **Table 4–28**. The impacts are calculated for two types of receptors: the general public living within 80 kilometers (50 miles) of F-Canyon in the year 2010, and a maximally exposed offsite individual (a member of the public assumed to be residing at the SRS site boundary and receiving the maximum dose). Since PUREX processing would produce radiological air emissions as well as radiological liquid effluent, doses to the public were calculated considering both the air emissions and liquid effluent. Primary contributors to public doses would be from tritium gas (assumed to be tritium oxide) and krypton-85, which together contribute over 95 percent of the total calculated doses. The doses resulting from liquid effluent were estimated from data provided in support of the SRS Spent Nuclear Fuel Management Final EIS (DOE 2000) (see Appendix E, Section E.4.3, for details). The doses and duration from decladding and cleaning blanket spent nuclear fuel and treatment of driver spent nuclear fuel at ANL-W would be similar to those presented for Alternative 2 in Section 4.4.4.1. To put the impacts into perspective, comparisons to natural background radiation levels are included in Table 4–28.

Table 4–28 Annual and Project Total Radiological Impacts to the Public From Operational Activities Under Alternative 3

Receptor	*PUREX Process Declad and Cleaned Blanket Spent Nuclear Fuel at SRS* [a, b, c]	*Declad and Clean Blanket Spent Nuclear Fuel and Electrometallurgically Treat Driver Spent Nuclear Fuel at ANL-W*
Population Within 80 Kilometers (50 Miles) in the Year 2010		
Collective dose (person-rem per year) [d]	0.020	0.0030
Excess latent cancer fatalities (per year)	0.000010	1.5×10^{-6}
Project total excess latent cancer fatalities [f]	0.000010	8.3×10^{-6}
Maximally Exposed Offsite Individual		
Dose (millirem per year) [d]	0.00051	0.00038
Percent of annual background radiation [e]	0.00017	0.00011
Latent cancer fatality risk (per year)	2.6×10^{-10}	1.9×10^{-10}
Project total lifetime cancer fatality risk [f]	2.6×10^{-10}	1.0×10^{-9}
Average Individual Within 80 Kilometers (50 Miles)		
Dose (millirem per year) [g]	0.000024	0.000012
Latent cancer fatality risk (per year)	1.2×10^{-11}	6.2×10^{-12}
Project total lifetime cancer fatality risk [f]	1.2×10^{-11}	3.5×10^{-11}

[a] Includes dose from air emissions and liquid effluent over the six-month processing duration.
[b] Doses to the population and the maximally exposed offsite individual from liquid effluent are 0.00068 person-rem and 0.00012 millirem, respectively.
[c] Since PUREX operations would last less than one year, the values of the project total dose and risk are equal to the corresponding annual values.
[d] Annual maximum dose during normal operations.
[e] The annual natural background radiation level at INEEL and at SRS is about 360 and 300 millirem, respectively, for the average individual (see Tables 3–8 and 3–20); the population within 80 kilometers (50 miles) in the year 2010 would receive 86,500 person-rem at INEEL and 254,000 person-rem at SRS.
[f] Total calculated risk over nine years at ANL-W and six months at SRS.
[g] Obtained by dividing the population dose by the number of people projected to live in the year 2010 within 80 kilometers (50 miles) of ANL-W (240,338) and SRS F-Canyon (848,000).

As shown in this table, the expected radiation doses to the maximally exposed offsite individual and the general public would be much smaller than the limit of 10 millirem per year set by the EPA (40 CFR 61) and DOE (DOE Order 5400.5).

Table 4–29 summarizes worker population doses. Occupational doses were estimated by examining the type and duration of various operations performed by SRS workers involved with the PUREX process. The estimated annual total worker population dose would be 75 person-rem, with an average individual dose of 500 millirem per year for each of the 150 involved workers. If these estimates were projected for six months of PUREX activities, the project total worker population dose would be 38 person-rem, leading to a risk of 0.015 latent cancer fatalities. The estimated annual total worker population dose to treat driver spent nuclear fuel at ANL-W is 22 person-rem, as indicated in Section 4.4.4.1.

Table 4–29 Annual and Project Total Radiological Impacts to Workers From Operational Activities Under Alternative 3

Receptor	*Impacts*	
Worker [a]	Operations at SRS	Operations at ANL-W
Average worker dose (millirem per year)	250 [b]	60
Average worker latent cancer fatality risk (project total)	0.00010 [b]	0.00022 [c]
Worker Population		
Collective dose (person-rem per year)	38 [b]	22
Excess latent cancer fatalities (per year)	0.015 [b]	0.0088
Project total dose (person-rem)	38 [b]	231 [c]
Project total excess latent cancer fatalities	0.015 [b]	0.092 [c]

[a] The regulatory dose limit for an individual worker is 5,000 millirem per year (10 CFR 835). However, the maximum annual dose to a worker would be kept below the DOE Administrative Control Level of 2,000 millirem per year, as established for all DOE activities in DOE Order N 441.1.
[b] Operations at SRS to treat blanket spent nuclear fuel at F-Canyon are performed over six months.
[c] Operations at ANL-W to declad and clean blanket spent nuclear fuel and treat driver spent nuclear fuel are performed over nine years plus one year for deactivation of processing facilities; see Section 4.4.1.

Sources: ANL 1999, DOE 2000.

As shown in Tables 4–28 and 4–29:

- The annual dose to the maximally exposed offsite individual at ANL-W would be 0.00038 millirem per year, with an associated risk of developing a fatal cancer of 1.9×10^{-10} per year (or one chance in 5.3 billion that the individual would develop a fatal cancer per year of exposure).

- The collective dose to the population within 80 kilometers (50 miles) of the ANL-W facilities would be 0.0030 person-rem per year, with an associated 1.5×10^{-6} latent cancer fatalities per year (or one chance in 670,000 that the population would experience a fatal cancer per year of exposure).

- The project total risk to the population within 80 kilometers (50 miles) of the ANL-W site would be 8.3×10^{-6} latent cancer fatalities (or one chance in 120,000 that the exposed population would experience a fatal cancer).

- The collective dose to ANL-W facility workers would be 22 person-rem per year, with an associated 0.0088 latent cancer fatalities per year (or one chance in 113 that the workers would experience a fatal cancer per year of operation).

- The project total dose to ANL-W facilities workers would be 231 person-rem, with an associated 0.092 latent cancer fatalities (or one chance in 11 that the exposed workers would experience a fatal cancer).

- The project total dose to the maximally exposed offsite individual at SRS from six-month PUREX processing would be 0.00051 millirem, with an associated risk of developing a fatal cancer of 2.6×10^{-10} (or one chance in 3.8 billion that the individual would develop a fatal cancer).

- The project total dose to the population within 80 kilometers (50 miles) of the F-Canyon would be 0.020 person-rem, with an associated 0.000010 latent cancer fatalities (or one chance in 100,000 that the exposed population would experience a fatal cancer).

- The project total dose to F-Canyon facility workers would be 38 person-rem, with an associated 0.015 latent cancer fatalities (or one chance in 67 that the exposed workers would experience a fatal cancer).

These results indicate that no increase in the expected number of fatal cancers among the various affected populations would result from the activities under this alternative.

Hazardous Chemical Impacts

The hazardous chemical impacts to the public residing at the INEEL site boundary from the operational activities at ANL-W under Alternative 3 would be similar to the impacts evaluated for Alternative 1 described in Section 4.3.4.1. The results indicate that no adverse toxic health or cancer effects would be expected from exposure to hazardous chemicals released under this alternative. The existing baseline chemical environment is presented in Section 3.2.10.2.

For SRS, both carcinogenic and noncarcinogenic health effects to the public were assessed from exposure to hazardous chemicals; the results are summarized in **Table 4–30**.

Table 4–30 Hazardous Chemical Impacts to the Public From Operational Activities at SRS Under Alternative 3

Chemical	*Annual Concentration (milligrams per cubic meter)*	*Hazard Quotient (noncarcinogenic chemicals)*	*Cancer Risk (carcinogenic chemicals)*
Benzene	1.4×10^{-6}	None	1.1×10^{-8}
Ethyl benzene	1.3×10^{-6}	1.3×10^{-6}	None
Formaldehyde	1.3×10^{-6}	None	1.6×10^{-8}
Hexane	1.4×10^{-6}	7.1×10^{-6}	None
Manganese	1.3×10^{-6}	0.025	None
Methyl ethyl ketone	2.5×10^{-6}	2.5×10^{-6}	None
Methylene chloride	7.1×10^{-7}	None	3.3×10^{-10}
Naphthalene	1.3×10^{-6}	0.00042	None
Toluene	1.4×10^{-6}	3.5×10^{-6}	None
Vinyl acetate	1.3×10^{-6}	6.3×10^{-6}	None
Hazard Index		0.025	Not applicable

The results indicate that no adverse toxic (noncarcinogenic) health effects or cancer potency are expected from exposure to hazardous chemicals released at SRS under this alternative. See Appendix E, Section E.5, for more details. The existing baseline chemical environment is presented in Section 3.3.10.2.

4.5.4.2 Facility Accidents

Radiological Impacts

Potential radiological impacts to the public and a noninvolved onsite worker due to accidents during operational activities associated with decladding, cleaning, and PUREX processing of blanket spent nuclear fuel and electrometallurgical treatment of driver spent nuclear fuel are summarized and presented in this section. The detailed analysis of facility accidents, with the associated assumptions, is presented in Appendix F. The detailed analysis considered a wide spectrum of potential accident scenarios, including fire, spills, criticality, earthquake, and aircraft crash. The aircraft crash and criticality events were determined to have an occurrence frequency of less than 10^{-7} per year, and consequence analyses for these two events were not performed. Decladding and cleaning of blanket spent nuclear fuel would be performed in the Hot Fuel Examination Facility; treatment of driver spent nuclear fuel would be performed in both the Hot Fuel Examination Facility and the Fuel Conditioning Facility. Because driver spent nuclear fuel processing would take place in both of these facilities, the beyond-design-basis earthquake event is assessed for the driver spent nuclear fuel, taking into account the multifacility impacts of this event. Decladding and cleaning blanket spent nuclear fuel would be performed only in the Hot Fuel Examination Facility. The multifacility impacts of the beyond-design-basis earthquake are not relevant to the blanket spent nuclear fuel. Therefore, only the higher frequency design-basis earthquake event was analyzed for the blanket spent nuclear fuel. **Table 4–31** presents the frequencies and consequences of the postulated set of accidents at ANL-W to the maximally exposed offsite individual, the offsite population residing within 80 kilometers (50 miles) of the facility, and a noninvolved worker located 100 meters (330 feet) to 230 meters (755 feet) from the facility.

Table 4–31 Accident Frequency and Consequences at ANL-W Under Alternative 3

Accident	*Frequency (event per year)*	*Maximally Exposed Offsite Individual*		*Population Within 80 Kilometers (50 Miles)*		*Noninvolved Worker*	
		Dose (millirem)	*Latent Cancer Fatality Risk* [a]	*Dose (person-rem)*	*Excess Latent Cancer Fatalities* [b]	*Dose (millirem)*	*Latent Cancer Fatality Risk* [a]
Driver Spent Nuclear Fuel							
Salt powder spill	0.01	0.00046	2.3×10^{-10}	0.000098	4.9×10^{-8}	4.7×10^{-7}	1.9×10^{-13}
Salt transfer drop	1.0×10^{-7}	0.19	9.5×10^{-8}	0.022	0.000011	0.073	2.9×10^{-8}
Transuranic waste fire	0.001	0.059	3.0×10^{-8}	0.0071	3.6×10^{-6}	0.22	8.8×10^{-8}
Cask drop	0.01	0.030	1.5×10^{-8}	0.0035	1.8×10^{-6}	0.00084	3.4×10^{-10}
Design-basis earthquake	0.008	12	6.0×10^{-6}	1.4	0.0007	4.7	1.9×10^{-6}
Beyond-design-basis earthquake	0.00001	22,000	0.022	2,500	1.3	370	0.00015
Blanket Spent Nuclear Fuel							
Cask drop	0.01	0.0024	1.2×10^{-9}	0.00028	1.4×10^{-7}	0.000049	2.0×10^{-11}
Transuranic waste fire	0.001	0.059	3.0×10^{-8}	0.0071	3.6×10^{-6}	0.22	8.8×10^{-8}
Sodium fire [c]	0.008	5.9	3.0×10^{-6}	0.69	0.00035	0.054	2.2×10^{-8}

[a] Increased likelihood of a latent cancer fatality.
[b] Increased number of latent cancer fatalities.
[c] The frequency for this accident is the frequency for the facility design-basis earthquake initiating cell fire.

The dose to the maximally exposed offsite individual was calculated for the 95th percentile meteorological conditions. The doses to the population and the noninvolved worker were calculated using 50th percentile meteorological conditions. DOE did not quantitatively estimate the involved worker dose due to accidents (see the discussion on the involved worker in Section 4.2.4.2). The accident risks for the same receptors are summarized in **Table 4–32**.

Table 4–32 Annual Cancer Risks Due to Accidents at ANL-W Under Alternative 3

Accident	*Maximally Exposed Offsite Individual* [a]	*Population Within 80 Kilometers (50 Miles)* [b]	*Noninvolved Worker* [a]
Driver Spent Nuclear Fuel			
Salt powder spill	2.3×10^{-12}	4.9×10^{-10}	1.9×10^{-15}
Salt transfer drop	9.5×10^{-15}	1.1×10^{-12}	2.9×10^{-15}
Transuranic waste fire	3.0×10^{-11}	3.6×10^{-9}	8.8×10^{-11}
Cask drop	1.5×10^{-10}	1.7×10^{-8}	3.4×10^{-12}
Design-basis earthquake	4.8×10^{-8}	5.6×10^{-6}	1.5×10^{-8}
Beyond-design-basis earthquake	2.2×10^{-7}	0.000013	1.5×10^{-9}
Blanket Spent Nuclear Fuel			
Cask drop	1.2×10^{-11}	1.4×10^{-9}	2.0×10^{-13}
Transuranic waste fire	3.0×10^{-11}	3.6×10^{-9}	8.8×10^{-11}
Sodium fire	2.4×10^{-8}	2.8×10^{-6}	1.7×10^{-10}

[a] Increased likelihood of a latent cancer fatality.
[b] Increased number of latent cancer fatalities.

For accidents at ANL-W, the highest risk of a latent cancer fatality to the maximally exposed offsite individual and to a noninvolved worker would be 2.2×10^{-7} per year (or one chance in 4.5 million that the individual would develop a fatal cancer per year of operation) and 1.5×10^{-8} per year (or one chance in 66.7 million that the individual would develop a fatal cancer per year of operation), respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.000013 per year (or one chance in 76,920 that the population would experience a fatal cancer per year of operation).

The potential radiological impacts to the public and a noninvolved onsite worker due to accidents during PUREX operational activities at SRS are summarized below. The detailed analysis of facility accidents, with the associated assumptions, is presented in Appendix F. **Table 4–33** presents the frequencies and consequences of the postulated set of accidents to the maximally exposed offsite individual, the offsite population residing within 80 kilometers (50 miles) of the facility, and a noninvolved worker located 100 meters (330 feet) to 350 meters (1150 feet) from the facility. The 350-meter (1150-foot) distance leads to a higher dose to the noninvolved worker for the scenarios with elevated releases.

The dose to the maximally exposed offsite individual was calculated for the 95th percentile meteorological conditions. The doses to the population and the noninvolved worker were calculated using 50th percentile meteorological conditions. DOE did not quantitatively estimate the facility worker population dose due to accidents. The accident risks for the same receptors are summarized in **Table 4–34**.

Table 4–33 Accident Frequency and Consequences at SRS Under Alternative 3

Accident	Frequency (event per year)	Maximally Exposed Offsite Individual		Population Within 80 Kilometers (50 Miles)		Noninvolved Worker	
		Dose (millirem)	Latent Cancer Fatality Risk [a]	Dose (person-rem)	Excess Latent Cancer Fatalities [b]	Dose (millirem)	Latent Cancer Fatality Risk [a]
Fire (F-Canyon)	0.000061	610	0.00031	5500	2.8	2300	0.00092
Explosion (FB-Line)	0.00010	6.5	3.3×10^{-6}	53	0.027	19	7.6×10^{-6}
Design-basis earthquake (F-Canyon)	0.00013	1100	0.00055	2100	1.1	12000	0.0048
Design-basis earthquake (FB-Line)	0.00013	58	0.000029	120	0.06	900	0.00036
Criticality	0.00010	11	5.5×10^{-6}	59	0.030	37	0.000015

[a] Increased likelihood of a latent cancer fatality.
[b] Increased number of latent cancer fatalities.

Table 4–34 Annual Cancer Risks Due to Accidents at SRS Under Alternative 3

Accident	Maximally Exposed Offsite Individual [a]	Population Within 80 Kilometers (50 Miles) [b]	Noninvolved Worker [a]
Fire (F-Canyon)	1.9×10^{-8}	0.00017	5.6×10^{-8}
Explosion (FB-Line)	3.3×10^{-10}	2.7×10^{-6}	7.6×10^{-10}
Design-basis earthquake (F-Canyon)	7.2×10^{-8}	0.00014	6.2×10^{-7}
Design-basis explosion (FB-Line)	3.8×10^{-9}	7.8×10^{-6}	4.7×10^{-8}
Criticality	5.5×10^{-10}	3.0×10^{-6}	1.5×10^{-9}

[a] Increased likelihood of a latent cancer fatality.
[b] Increased number of latent cancer fatalities.

For accidents at SRS, the highest risk of a latent cancer fatality to the maximally exposed offsite individual and to a noninvolved worker would be 7.2×10^{-8} per year (or one chance in 13.9 million that the individual would develop a fatal cancer per year of operation) and 6.2×10^{-7} per year (or one chance in 1.6 million that the individual would develop a fatal cancer per year of operation), respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.00017 per year (or one chance in 5,880 that the population would experience a fatal cancer per year of operation).

Hazardous Chemical Impacts

Nonradiological impacts are evaluated in terms of comparison to ERPG. ERPG values are estimates of airborne concentration thresholds above which one can reasonably anticipate observing adverse effects (see Appendix F, Section F.3.1.2, for details).

The hazardous chemical impacts of potential facility accidents at ANL-W associated with the treatment of driver spent nuclear fuel using electrometallurgical treatment are summarized in **Table 4–35**.

Table 4–35 Hazardous Chemical Impacts Due to Accidents at ANL-W Under Alternative 3

Accident	*Frequency (event per year)*	*Receptor*	*Exposure*
Uranium handling accident	0.01	Noninvolved worker	Uranium: less than ERPG-1
		Maximally exposed offsite individual	Uranium: less than ERPG-1
Design-basis earthquake	0.0002	Noninvolved worker	Uranium: less than ERPG-1
		Maximally exposed offsite individual	Uranium: less than ERPG-1
Sodium fire	0.008	Noninvolved worker	Sodium: less than ERPG-1
		Maximally exposed offsite individual	Sodium: less than ERPG-1
Uranium fire	0.00001	Noninvolved worker	Uranium: less than ERPG-1
		Maximally exposed offsite individual	Uranium: less than ERPG-1
Beyond-design-basis earthquake	0.00001	Noninvolved worker	Cadmium: less than ERPG-1 Uranium: less than ERPG-1
		Maximally exposed offsite individual	Cadmium: less than ERPG-1 Uranium: less than ERPG-1

ERPG = Emergency Response Planning Guideline.

The SRS Spent Nuclear Fuel Management Final EIS (DOE 2000) analyzed the consequences of three chemical spills involving hazardous chemicals in the F-Area: (1) the loss of 50 percent sodium hydroxide containment from a skid-mounted 1,000-gallon dumpster; (2) the loss of 50 percent nitric acid containment from a skid-mounted 1,000-gallon dumpster; and (3) the loss of 30 percent sodium nitrite containment from a skid-mounted 1,000-gallon dumpster and an adjacent 1,600-gallon holdup tank. These analyses are summarized in the **Table 4–36**, and are considered representative of wet storage accidents at SRS.

Table 4–36 Hazardous Chemical Impacts Due to Accidents at SRS Under Alternative 3

Accident	*Frequency (Event Per Year)*	*Receptor*	*Exposure*
Wet storage, container rupture	0.005	Noninvolved worker	Sodium hydroxide: less than Permissible Exposure Limit-Time Weighted Average
Wet storage, container rupture	0.005	Noninvolved worker	Nitric acid: less than Permissible Exposure Limit-Time Weighted Average
		Maximally exposed offsite individual	Nitric acid: less than Permissible Exposure Limit-Time Weighted Average
Wet storage, container rupture	0.006	Noninvolved worker	Sodium nitrite: less than Permissible Exposure Limit-Time Weighted Average

Permissible Exposure Limit-Time Weighted Average is used for chemicals having no ERPG values. It is considered to be less than ERPG-1.

Source: DOE 2000.

4.5.5 Environmental Justice

As discussed in Appendix H, Executive Order 12898 directs Federal agencies to address disproportionately high and adverse health or environmental effects of alternatives on minority or low-income populations.

Analyses of normal operations and accident conditions presented in Section 4.3.4 show the risk of latent cancer fatalities to the public residing within 80 kilometers (50 miles) of the electrometallurgical treatment and decladding and cleaning processing facilities at ANL-W and the PUREX processing facility at SRS to

be much lower than 1. Therefore, radiological and nonradiological risks posed by implementation of this alternative would not result in disproportionately high and adverse consequences to any particular group within the general population, including minority or low-income populations.

4.5.6 Waste Management

ANL-W

This alternative would generate process waste from treatment operations, other associated process waste from normal support operations, and deactivation waste following the conclusion of operations. Process waste would include fuel assembly hardware and high-level radioactive metallic and ceramic waste. Other associated process waste would include operational waste such as failed equipment, rags, packaging materials, and other miscellaneous items. Deactivation waste would include the disposal of process equipment and other materials. All of these materials would be categorized according to existing DOE Orders and ANL-W waste management procedures. The anticipated categorization of these waste types generated at ANL-W and their expected interim storage and final disposal locations are given in Table 4–9 (see Section 4.2.6). The quantities of various waste forms generated as a result of Alternative 3 at ANL-W are provided in **Table 4–37**.

Estimates of the total amount of other associated process waste that would be generated are based on an evaluation of waste forecasts from ANL-W that accounted only for the fraction of total ANL-W waste that would be attributable to the treatment of sodium-bonded spent nuclear fuel. The values in Table 4–37 are for disposal and account for volume reduction. It is anticipated that a large fraction of the low-level radioactive waste generated as a result of Alternative 3 could be volume-reduced at the Waste Experimental Reduction Facility at INEEL, prior to disposal at the INEEL Subsurface Disposal Area at the Radioactive Waste Management Complex.

Table 4–37 Amounts of Waste Generated at ANL-W Under Alternative 3[a]

Waste Stream	*Waste Quantities*	
	Volume (cubic meters)	*Mass (kilograms)*
Direct Process Waste		
Fuel assembly hardware (low-level radioactive waste)	37.5	13,100
High-level radioactive ceramic waste	16.3 (26 canisters)[b]	24,400
High-level radioactive metallic waste	1.3 (2 canisters)[b]	2,500
Spent nuclear fuel	0	0
Other Associated Process Waste		
High-level radioactive waste	0.4 (1 canister)[b]	220
Low-level radioactive waste[c]	555	113,000
Transuranic waste	9.1	3,800
Mixed waste	27.5	14,800
Sanitary waste	4,960	1.72×10^{6}
Deactivation Waste		
Low-level radioactive waste[c]	178	60,000
Transuranic waste	1.6	853
Mixed waste	5.1	3,400

[a] These waste generation estimates are through the year 2015. This is the assumed date that these materials might be sent to the repository. Treatment, high-level radioactive waste processing, deactivation, and interim storage would be accomplished during this time period.

[b] Standardized canisters.

[c] The volumes listed represent final disposal volumes following volume reduction at the Waste Experimental Reduction Facility at INEEL.

Source: ANL 1999.

The waste values in Table 4–37 are total quantities that would be produced as a result of Alternative 3 operations at ANL-W. They are not incremental increases over the volumes provided in Table 4–10 that would result from the No Action Alternative. In Alternative 3, the driver spent nuclear fuel would be transformed into high-level radioactive waste forms (ceramic and metallic) at ANL-W for disposal in the repository, and in this conversion process, the total volume of material to be disposed of in the repository would be reduced from direct disposal values. The blanket spent nuclear fuel would be cleaned and declad and sent to SRS for PUREX processing. The high-level radioactive waste that would be generated from PUREX processing at SRS is presented in Table 4–39.

Direct Process Waste

For this alternative, fuel hardware would be removed from the fuel elements in the Fuel Conditioning Facility air cell and disposed of as low-level radioactive waste. These components are primarily stainless steel materials that contain short-lived radionuclides. This waste stream has been produced at ANL-W for many years and would be handled, as in the past, according to DOE Orders and ANL-W waste management procedures. In addition, the blanket spent nuclear fuel cladding is included in the fuel hardware stream.

Under this alternative, metallic and ceramic high-level radioactive waste would be a primary product of the electrometallurgical treatment of driver spent nuclear fuel. The salt removed from the electrorefiners would contain the majority of fission products and transuranics from the spent nuclear fuel. This removed salt would be packaged and transferred to the Hot Fuel Examination Facility for processing into ceramic waste. The metallic waste form would consist primarily of stainless steel cladding hulls containing the noble metal fission products. The hulls would be removed from the electrorefiner and packaged for shipment to the Hot Fuel Examination Facility for processing into the metallic waste form. Both the ceramic and metallic waste would be categorized as high-level radioactive waste. The volumes of waste forms provided in Table 4–37 are for the standardized canisters required for disposal of these materials.

The metallic and ceramic high-level radioactive waste that would be generated as a result of electrometallurgical treatment of driver spent nuclear fuel under this alternative and the driver and blanket spent nuclear fuel generated during the demonstration project at ANL-W would be stored temporarily at the Radioactive Scrap and Waste Facility at ANL-W to allow retrieval for future disposal. The Radioactive Scrap and Waste Facility was designed and constructed for temporary storage of this type of waste. Shielding would be provided by a combination of (1) steel storage liners which would store the waste, and (2) the soil surrounding the liners. When a geologic repository is available, the waste cans containing the metallic and ceramic high-level radioactive waste would be removed from storage, shipped to the INEEL Dry Transfer Facility, and prepared for shipment to the repository.

Other Associated Process High-Level Radioactive Waste

High-level radioactive waste could be generated as a result of blanket spent nuclear fuel processing at ANL-W. This would result from activities in the Hot Fuel Examination Facility. Material in this waste stream would consist of the absorbent used in the off-gas system which has collected the volatile radionuclides released from the spent nuclear fuel when heated.

The volume of high-level radioactive waste that would be generated is expected to be less than the amount needed to fill a single high-level radioactive waste canister. Conservatively the volume of a single canister, 0.4 cubic meters, has been used for the volume of high-level radioactive waste generated.

Other Associated Process Low-Level Radioactive Waste

Low-level radioactive waste would be generated as a result of decladding and cleaning blanket spent nuclear fuel and treatment of driver spent nuclear fuel at ANL-W. This would result from activities in the Fuel Conditioning Facility and the Hot Fuel Examination Facility at ANL-W (e.g., equipment decontamination and repair), as well as in other facilities at ANL-W (e.g., analytical laboratory activities). Material in this waste stream has been generated and routinely handled at ANL-W for many years.

The volume of low-level radioactive waste resulting from decladding and cleaning blanket spent nuclear fuel and electrometallurgical treatment of driver spent nuclear fuel at ANL-W that would require disposal (after volume reduction) would be approximately 50 cubic meters (1,766 cubic feet) per year. This represents approximately 1 percent of the total annual volume of low-level radioactive waste currently being disposed of at the INEEL Subsurface Disposal Area at the Radioactive Waste Management Complex, and the total of 555 cubic meters (19,600 cubic feet) represents approximately 0.5 percent of the total Radioactive Waste Management Complex disposal inventory.

Other Associated Process Transuranic Waste

Transuranic waste would be generated by Alternative 3 from decontamination activities for repair and maintenance of items, and miscellaneous work associated with processing the sodium-bonded spent nuclear fuel. Transuranic waste would be generated primarily from activities conducted in gloveboxes and hot cells at ANL-W.

All of the transuranic waste generated would be packaged and certified in accordance with Waste Isolation Pilot Plant Acceptance Criteria prior to transport to the Waste Isolation Pilot Plant facility. The transuranic waste that would be generated would amount to approximately 1 cubic meter (35 cubic feet) per year, which is less than 0.002 percent of the volume of transuranic waste in retrievable storage at the Radioactive Waste Management Complex at INEEL. The total volume of transuranic waste is 9.1 cubic meters (321 cubic feet), which is less than 0.005 percent of the estimated total volume of transuranic waste to be placed at the Waste Isolation Pilot Plant.

Other Associated Process Mixed Waste

Mixed waste would be generated primarily from the disposal of any cadmium-contaminated equipment or cleanup material and the analysis of cadmium samples. Mixed waste would be handled according to ANL-W procedures that require limited accumulation at the point of generation. Interim storage of this waste would be accomplished at the Radioactive Scrap and Waste Facility prior to eventual disposal. The Radioactive Scrap and Waste Facility is a permitted mixed waste storage facility for these materials. The mixed waste streams that contribute to the overall mixed waste generated by electrometallurgical treatment have been identified in the INEEL Site Treatment Plan (DOE 1995b).

Deactivation Waste

A variety of waste would be generated as part of deactivation activities associated with decladding and cleaning blanket spent nuclear fuel and the treatment of driver spent nuclear fuel at ANL-W. This would include process equipment and process material, such as electrorefiner cadmium from electrometallurgical treatment of driver spent nuclear fuel. Generated waste categories would include low-level radioactive waste, transuranic waste, and mixed waste. This waste would be categorized and disposed of according to DOE Orders and ANL-W radioactive waste management procedures, as described above for each waste category.

The largest volume of deactivation waste would be low-level radioactive waste generated as a result of equipment dismantling and disposal. Components of electrometallurgical treatment of the driver spent nuclear fuel that would require disposition include two electrorefiners, two hot isostatic presses, and two V-mixers, as well as other components such as the grinder/crusher. Decontamination of these components would generate additional mixed, transuranic, and low-level radioactive waste that would require management. The deactivation waste volume is generated over two years. The total deactivation waste would represent an additional 30 percent over the total associated process waste (excluding sanitary waste) requiring disposal.

SRS

The PUREX process at SRS would generate process waste from treatment operations and other associated process waste from support operations. Process waste would include high-level radioactive waste. Other associated process waste would include operational waste such as failed equipment, rags, packaging materials, and other miscellaneous items. The associated process waste includes low-level radioactive waste, transuranic waste, and mixed waste. All of the waste streams would be categorized according to existing DOE Orders and SRS waste management procedures. The anticipated categorization of the waste types and their expected interim storage and final disposal locations are given in **Table 4–38**.

Table 4–38 Waste Material Categories at SRS and Interim and Final Locations

Waste Stream	*Category*	*Interim Storage Location*	*Final Disposal Location*
Process Waste			
Liquid waste form	High-level radioactive waste	Initial storage in the high-level radioactive waste Tank Farm followed by post-process storage at the Defense Waste Processing Facility.	Geologic repository
Other Associated Process Waste			
Less than 100 nanocuries per gram transuranic waste	Low-level radioactive waste	None	Low-activity waste vaults
Greater than 100 nanocuries per gram transuranic waste	Transuranic waste	Transuranic waste storage pads	Waste Isolation Pilot Plant
Contaminated	Mixed waste	Mixed waste storage buildings	Offsite

Estimates of the amounts of waste that would be generated as a result of the PUREX processing at SRS are provided in **Table 4–39**. These values are based on an evaluation of waste forecasts that account only for the fraction of total waste that would be attributable to processing the blanket spent nuclear fuel pins.

As indicated in the following waste type discussions, the amounts of waste associated with this alternative are relatively small compared to onsite and offsite management capacities.

Direct Process Waste

During the PUREX process, liquid high-level radioactive waste would be produced (along with plutonium metal and uranium solution). The liquid waste would be processed in the Defense Waste Processing Facility to yield vitrified high-level radioactive waste (borosilicate glass) and saltstone. This high-level radioactive waste would be temporarily stored at the Defense Waste Processing Facility pending ultimate disposal in a geologic repository. The saltstone is a cement form low-level radioactive waste that is generated or a by-product of SRS tank farm operations. The saltstone would be disposed of on site in the Z-Area Saltstone Vaults. The volume of this saltstone would be about 0.12 percent of the 1.11 million cubic meters (39.2 million cubic feet) storage capacity of the vaults.

Table 4–39 Amounts of Waste Generated at SRS Under Alternative 3

Waste Stream	*Waste Quantities (cubic meters)* [a]
Direct Process Waste	
Liquid high-level radioactive waste	510
Solid high-level radioactive waste [b, c]	5.6 (9 canisters) [b]
Saltstone [c]	1,290
Other Associated Process Waste	
Low-level radioactive waste	900 [d]
Transuranic waste	90
Mixed waste	6.9

[a] These values are estimated based on the heavy metal mass ratio of similar materials processed at SRS (20 metric tons of heavy metal) and provided in DOE 2000.
[b] Standardized high-level radioactive waste (Defense Waste Processing Facility) canisters.
[c] These waste forms result from processing the liquid high-level radioactive waste.
[d] Final disposal volume following a volume reduction (a reduction factor of 4 was assumed).

Other Associated Process Low-Level Radioactive Waste

Low-level radioactive waste would be generated during the PUREX process. The volume of low-level radioactive waste resulting from this alternative (after volume reduction) would be about 3 percent of the total 30,500-cubic meter (1.08 million-cubic foot) disposal capacity of the low-activity waste vaults.

Other Associated Process Transuranic Waste

The volume of transuranic waste that would be generated during the PUREX process would be only about 0.05 percent of the current 168,500-cubic meter (5.95 million-cubic foot) limit for the Waste Isolation Pilot Plant (DOE 1997).

Other Associated Process Mixed Waste

Mixed waste that would be generated during the PUREX process would be temporarily stored on site in the Mixed Waste Storage Buildings prior to eventual offsite disposal. The volume of this waste would be about 0.36 percent of the 1,900-cubic meter (67,100-cubic foot) storage capacity of these storage buildings.

4.6 ALTERNATIVE 4: MELT AND DILUTE BLANKET FUEL AND ELECTROMETALLURGICALLY TREAT DRIVER FUEL AT ANL-W

Under this alternative, the sodium-bonded blanket spent nuclear fuel would be cleaned to remove metallic sodium and then treated using the melt and dilute process at ANL-W. The melt and dilute product from treatment of this fuel would be stored at the Radioactive Scrap and Waste Facility pending repackaging and transportation for disposal in a geologic repository. The removed sodium would be stabilized using an oxidation/carbonation process (ANL 1999). The sodium-bonded driver spent nuclear fuel would be treated at ANL-W using the electrometallurgical treatment process. The process steps for the electrometallurgical treatment of driver spent nuclear fuel would be similar to those described earlier in Section 4.3 and in Appendix C. The treatment of driver spent nuclear fuel using the electrometallurgical process could start as early as 2000 and could be completed by 2006 to 2007. The preparation of blanket spent nuclear fuel could start in 2003, and subsequent melt and dilute treatment at ANL-W could start in 2005 and could be completed by 2012. In addition, a full year of operations would be needed to deactivate the processing equipment and establish an industrially safe configuration.

4.6.1 Air Quality

Nonradiological Gaseous Emissions

It is expected that criteria and hazardous air pollutants released from operational activities at ANL-W under this alternative would be the same as for Alternative 1, as described in Section 4.3.1 (see also Appendix E, Section E.5.3.1, for more detail). Baseline air quality concentrations are presented in Section 3.2.3.1.

Radiological Gaseous Emissions

Cleaning blanket spent nuclear fuel and electrometallurgically treating driver spent nuclear fuel would release gaseous fission products to the argon cell environment. Krypton-85 and elemental tritium are the most prevalent radioactive gaseous fission products that would be released to the environment. The tritium released into the cell would not be oxidized due to a very low presence of oxygen and humidity in the argon cell. The argon cell also contains an equilibrium concentration of other radionuclide isotopes. Appendix E, Section E.4.1, provides a list of various isotopes that are present in the argon cell in nanocuries (10^{-9} curies) and are released to the atmosphere through the facility stack, along with krypton-85 and elemental tritium. The maximum annual release of radioactive gaseous emissions would occur when electrometallurgical treatment processing of driver spent nuclear fuel is performed simultaneously with cutting blanket spent nuclear fuel for sodium removal prior to the melt and dilute process. This simultaneous operation would occur over a three-year period during the estimated 10 years of operation starting in 2003. Appendix E, Section E.4.2, provides details on releases during the processing period at ANL-W. Based on an annual blanket spent nuclear fuel processing (e.g., chopping and cleaning) throughput of 10 metric tons of heavy metal and electrometallurgical treatment processing of about 0.6 metric tons of heavy metal of driver spent nuclear fuel elements, about 809 curies of elemental tritium and 11,860 curies of gaseous krypton-85 could be released annually to the atmosphere (see Appendix E, Section E.4.2). The radiological exposures to the public and workers from radioactive emissions are presented in Section 4.6.4.

4.6.2 Water Resources

Surface Water

No surface water is used at ANL-W. Flood waters from the Big Lost River would not be expected to reach the facilities at ANL-W, as shown in Figure 3–3.

Nonradiological Liquid Effluent

There are no discharges to the surface waters at ANL-W, except for discharges of nonhazardous liquid waste to the sewage pond and to the industrial waste pond. Big Lost River, Little Lost River, and Birch Creek would not be impacted by activities associated with electrometallurgical and melt and dilute treatment processes. Current operating and monitoring practices would continue for stormwater and liquid effluent discharges associated with facilities at ANL-W (see Section 4.2.2).

During fuel treatment and associated activities, some hazardous materials may be used inside buildings. To prevent potential releases to surface or subsurface waters resulting from spills of hazardous materials used in buildings, these facilities are designed, constructed, and maintained to contain these materials. Double-contained pipes, leak detection, and secondary containment of tanks are some of the features used to prevent hazardous material releases to the environment. Following existing written procedures, spill containment and cleanup equipment is present in areas where hazardous materials are stored or used (DOE 1996b).

Radiological Liquid Effluent

No radiological liquid effluent generated by electrometallurgical and melt and dilute treatment process operations would be discharged to surface water.

Groundwater

Under this alternative at ANL-W, there would be little change in groundwater consumption for domestic use since there is little change expected in the number of workers. Water consumption for the electrometallurgical and melt and dilute treatment process operations would not impact the current water usage at ANL-W. The current water usage at ANL-W is 188 million liters (49.6 million gallons) per year.

Nonradiological Liquid Effluent

No nonradiological liquid effluent generated by electrometallurgical and melt and dilute treatment process operations would be discharged to groundwater.

Radiological Liquid Effluent

No radiological liquid effluent generated by electrometallurgical and melt and dilute treatment process operations would be discharged to groundwater.

4.6.3 Socioeconomics

Under this alternative, the existing facilities at ANL-W would remain operational. No new employment or in-migration of workers would be required. Thus, there would be no additional impacts on the socioeconomic conditions in the region around ANL-W and INEEL.

4.6.4 Public and Occupational Health and Safety

The assessments of potential incremental radiological and chemical impacts associated with this alternative are presented in this section. Summaries of radiological impacts from normal operations are presented in Tables 4–40 and 4–41 for the public and workers, respectively. The radiological impacts from a spectrum of hypothetical accident scenarios are provided in Tables 4–42 and 4–43. The impacts from hazardous chemical releases during accident conditions are presented in Table 4–44. Background information on the effects of radiation on human health and safety is presented in Appendix E, Section E.2.

4.6.4.1 Normal Operations

Radiological Impacts

Under this alternative, radioactive releases would occur during sodium-bonded blanket spent nuclear fuel cleaning and driver spent nuclear fuel chopping and electrorefining. All of these activities are performed in the argon cell. Appendix E, Sections E.3, E.4.1, and E.4.2, details the method and assumptions used for calculating the impacts of normal operational radiological releases on the public health and safety. The maximum annual dose to the public would result when both blanket and driver spent nuclear fuel would be treated simultaneously under this alternative. Appendix E, Section E.4.2, provides details on the treatment process duration and throughputs for each fuel type.

Calculated maximum annual and project total radiological impacts to the public are given in **Table 4–40**. The impacts are calculated for two types of receptors: the general public living within 80 kilometers

(50 miles) of ANL-W in the year 2010, and a maximally exposed offsite individual (a member of the public assumed to be residing at the INEEL site boundary and receiving the maximum dose). Primary contributors to doses to members of the public are releases of tritium gas (about 1 percent of which was conservatively assumed to be in oxidized form) and krypton-85, which together contribute over 99.9 percent of the total calculated doses. To put the operational impacts into perspective, comparisons with impacts from natural background radiation also are included in the table. As shown in this table, the expected radiation doses to the maximally exposed offsite individual and the general public are much smaller than the limit of 10 millirem per year set by the EPA (40 CFR 61) and DOE (DOE Order 5400.5).

Table 4–40 Annual and Project Total Radiological Impacts to the Public From Operational Activities Under Alternative 4

Receptor	*Electrometallurgically Treat Driver Spent Nuclear Fuel*	*Clean and Melt and Dilute Blanket Spent Nuclear Fuel*	*Total*
Population Within 80 Kilometers (50 Miles) in the Year 2010			
Collective dose (person-rem per year) [a]	0.0027	0.00028	0.0030
Excess latent cancer fatalities (per year)	1.4×10^{-6}	1.4×10^{-7}	1.5×10^{-6}
Project total excess latent cancer fatalities [b]	8.0×10^{-6}	3.4×10^{-7}	8.3×10^{-6}
Maximally Exposed Offsite Individual			
Dose (millirem per year) [a]	0.00033	0.000048	0.00038
Percent of annual background radiation [c]	0.000092	0.000013	0.00011
Latent cancer fatality risk (per year)	1.7×10^{-10}	2.4×10^{-11}	1.9×10^{-10}
Project total lifetime cancer fatality risk [b]	9.4×10^{-10}	5.8×10^{-11}	1.0×10^{-9}
Average Individual Within 80 Kilometers (50 Miles)			
Dose (millirem per year) [d]	0.000011	1.2×10^{-6}	0.000012
Latent cancer fatality risk (per year)	5.6×10^{-12}	5.8×10^{-13}	6.2×10^{-12}
Project total lifetime cancer fatality risk [b]	3.3×10^{-11}	1.4×10^{-12}	3.5×10^{-11}

[a] Annual maximum dose during normal operations.
[b] Total calculated risk over 13 years. Majority of the dose is a result of 9 years of operation, electrometallurgically treating driver spent nuclear fuel, and cleaning of blanket spent nuclear fuel. Limited offsite consequences are associated with the melt and dilute processing of cleaned spent nuclear fuel.
[c] The annual natural background radiation level at INEEL is about 360 millirem for the average individual (see Table 3–8); the population within 80 kilometers (50 miles) in the year 2010 would receive 86,500 person-rem.
[d] Obtained by dividing the population dose by the number of people projected to live within 80 kilometers (50 miles) of ANL-W in the year 2010 (240,338).

Table 4–41 summarizes worker population doses. Occupational doses were estimated by examining the type and duration of various operations performed by ANL-W workers involved with sodium-bonded spent nuclear fuel electrometallurgical and melt and dilute treatment processes. It was concluded that the average worker dose would not be different from what currently is being experienced. The estimated annual collective worker dose would be 22 person-rem, with an average individual dose of 60 millirem per year for each of the 346 involved workers. If these estimates were extended over the 13 years of treatment activities, and the dose (33 person-rem) from one year of deactivation activities was incorporated, the project total worker population dose would be 319 person-rem, leading to a risk of 0.13 latent cancer fatalities.

Table 4–41 Annual and Project Total Radiological Impacts to Workers From Operational Activities Under Alternative 4

Receptor	*Impacts*
Worker [a]	
Average worker dose (millirem per year)	60
Average worker latent cancer fatality risk (project total over 13 years)	0.00031
Worker Population	
Collective dose (person-rem per year) [b]	22
Excess latent cancer fatalities (per year)	0.0088
Project total dose (person-rem)	319
Project excess total latent cancer fatalities	0.13

[a] The regulatory dose limit for an individual worker is 5,000 millirem per year (10 CFR 835). However, the maximum annual dose to an involved worker would be kept below the DOE Administrative Control Level of 2,000 millirem per year, as established for all DOE activities in DOE Order N 441.1.

[b] The worker dose during one year of facility deactivation would be 33 person-rem.

Source: ANL 1999.

As shown in Tables 4–40 and 4–41:

- The annual dose to the maximally exposed offsite individual would be 0.00038 millirem per year, with an associated risk of developing a fatal cancer of 1.9×10^{-10} per year (or one chance in 5.3 billion that the individual would develop a fatal cancer per year of exposure).

- The collective dose to the population within 80 kilometers (50 miles) of the ANL-W facilities would be 0.0030 person-rem per year, with an associated 1.5×10^{-6} latent cancer fatalities per year (or one chance in 670,000 that the population would experience a fatal cancer per year of exposure).

- The project total risk to the population within 80 kilometers (50 miles) of the ANL-W facilities would be 8.3×10^{-6} latent cancer fatalities (or one chance in 120,000 that the exposed population would experience a fatal cancer).

- The collective dose to facility workers would be 22 person-rem per year, with an associated 0.0088 latent cancer fatalities per year (or one chance in 113 that the workers would experience a fatal cancer per year of operation).

- The project total dose to facility workers would be 319 person-rem, with an associated 0.13 latent cancer fatalities (or one chance in eight that the exposed workers would experience a fatal cancer).

These results indicate that no increase in the expected number of fatal cancers among the various affected populations would result from the activities under this alternative.

Hazardous Chemical Impacts

The hazardous chemical impacts to the public residing at the INEEL site boundary from the operational activities at ANL-W under Alternative 4 would be similar to the impacts evaluated for Alternative 1, as described in Section 4.3.4.1. The results indicate that no adverse toxic health or cancer effects would be expected from exposure to hazardous chemicals released under this alternative. The existing chemical environment is presented in Section 3.2.10.2.

4.6.4.2 Facility Accidents

Radiological Impacts

Potential radiological impacts to the public and a noninvolved onsite worker due to accidents during operational activities associated with cleaning (sodium removal) blanket spent nuclear fuel for melt and dilute processing and treating driver spent nuclear fuel using electrometallurgical treatment are summarized and presented in this section. The detailed analysis of facility accidents, with their associated assumptions, is presented in Appendix F. The detailed analysis considered a wide spectrum of potential accident scenarios, including fire, spills, criticality, earthquake, and aircraft crash. The aircraft crash and criticality events were determined to have an occurrence frequency of less than 10^{-7} per year and consequence analyses for these two events were not performed. Processing of blanket spent nuclear fuel would be performed in the Hot Fuel Examination Facility; treatment of driver spent nuclear fuel would be performed in both the Hot Fuel Examination Facility and the Fuel Conditioning Facility. Because driver spent nuclear fuel processing would take place in both of these facilities, the beyond-design-basis earthquake event is assessed for the driver spent nuclear fuel, taking into account the multifacility impacts of this event, and releases from both the Hot Fuel Examination Facility and the Fuel Conditioning Facility from the single earthquake event. The melt and dilute processing of blanket spent nuclear fuel would be performed only in the Hot Fuel Examination Facility. Melt and dilute processing of the fuel could result in a greater number of accidents to be considered (waste processing-related events) in the assessment of accidents involving blanket spent nuclear fuel at ANL-W than declad and clean operations. The multifacility impacts of the beyond-design-basis earthquake are not relevant to the blanket spent nuclear fuel melt and dilute processing which occurs in only one facility. Therefore, only the higher frequency design-basis earthquake event was analyzed. **Table 4–42** presents the frequencies and consequences of the postulated set of accidents to the maximally exposed offsite individual, the offsite population residing within 80 kilometers (50 miles) of the facility, and a noninvolved worker located 100 meters (330 feet) to 230 meters (755 feet) from the facility. The 230-meter (755-foot) distance is the ANL-W bus staging area, which leads to a higher dose to the noninvolved worker for the scenarios with elevated releases.

The dose to the maximally exposed offsite individual was calculated for the 95th percentile meteorological conditions. The doses to the population and the noninvolved worker were calculated using 50th percentile meteorological conditions. DOE did not quantitatively estimate the involved worker dose due to accidents (see the discussion on the involved worker in Section 4.2.4.2). The accident risks for the same receptors are summarized in **Table 4–43**.

For accidents at ANL-W, the highest risk of a latent cancer fatality to the maximally exposed offsite individual and to a noninvolved worker would be 1.9×10^{-6} per year (or one chance in 526,300 that the individual would develop a fatal cancer per year of operation) and 4.9×10^{-8} per year (or one chance in 20.4 million that the individual would develop a fatal cancer per year of operation), respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.00022 per year (or one chance in 4,545 that the population would experience a fatal cancer per year of operation).

Hazardous Chemical Impacts

Nonradiological impacts are evaluated in terms of comparison to ERPG. ERPG values are estimates of airborne concentration thresholds above which one can reasonably anticipate observing adverse effects (see Appendix F, Section F.3.1.2, for details).

The nonradiological impacts of potential facility accidents (hazardous chemical) associated with the treatment of driver spent nuclear fuel using the electrometallurgical treatment process are summarized in **Table 4–44**.

Table 4–42 Accident Frequency and Consequences Under Alternative 4

Accident	Frequency (event per year)	Maximally Exposed Offsite Individual		Population Within 80 Kilometers (50 Miles)		Noninvolved Worker	
		Dose (millirem)	Latent Cancer Fatality Risk [a]	Dose (person-rem)	Excess Latent Cancer Fatalities [b]	Dose (millirem)	Latent Cancer Fatality Risk [a]
Driver Spent Nuclear Fuel							
Salt powder spill	0.01	0.00046	2.3×10^{-10}	0.000098	4.9×10^{-8}	4.7×10^{-7}	1.9×10^{-13}
Salt transfer drop	1.0×10^{-7}	0.19	9.5×10^{-8}	0.022	0.000011	0.073	2.9×10^{-8}
Transuranic waste fire	0.001	0.059	3.0×10^{-8}	0.0071	3.6×10^{-6}	0.22	8.8×10^{-8}
Cask drop	0.01	0.030	1.5×10^{-8}	0.0035	1.8×10^{-6}	0.00084	3.4×10^{-10}
Design-basis earthquake	0.008	12	6.0×10^{-6}	1.4	0.0007	4.7	1.9×10^{-6}
Beyond-design-basis earthquake	0.00001	22,000	0.022	2,500	1.3	370	0.00015
Blanket Spent Nuclear Fuel							
Cask drop	0.01	0.0024	1.2×10^{-9}	0.00028	1.4×10^{-7}	0.000049	2.0×10^{-11}
Transuranic waste fire	0.001	0.059	3.0×10^{-8}	0.0071	3.6×10^{-6}	0.22	8.8×10^{-8}
Sodium fire [c]	0.008	5.9	3.0×10^{-6}	0.69	0.00035	0.054	2.2×10^{-8}
Design-basis earthquake	0.008	471	0.00024	56.1	0.028	15.2	6.1×10^{-6}
Waste handling spill	0.0024	15	7.5×10^{-6}	1.8	0.00090	0.49	2.0×10^{-7}

[a] Increased likelihood of a latent cancer fatality.
[b] Increased number of latent cancer fatalities.
[c] The frequency of this accident is the frequency of the facility design-basis earthquake initiating a cell fire.

Table 4–43 Annual Cancer Risks Due to Accidents at ANL-W Under Alternative 4

Accident	Maximally Exposed Offsite Individual [a]	Population Within 80 Kilometers (50 Miles) [b]	Noninvolved Worker [a]
Driver Spent Nuclear Fuel			
Salt powder spill	2.3×10^{-12}	4.9×10^{-10}	1.9×10^{-15}
Salt transfer drop	9.5×10^{-15}	1.1×10^{-12}	2.9×10^{-15}
Transuranic waste fire	3.0×10^{-11}	3.6×10^{-9}	8.8×10^{-11}
Cask drop	1.5×10^{-10}	1.7×10^{-8}	3.4×10^{-12}
Design-basis earthquake	4.8×10^{-8}	5.6×10^{-6}	1.5×10^{-8}
Beyond-design-basis earthquake	2.2×10^{-7}	0.000013	1.5×10^{-9}
Blanket Spent Nuclear Fuel			
Cask drop	1.2×10^{-11}	1.4×10^{-9}	2.0×10^{-13}
Transuranic waste fire	3.0×10^{-11}	3.6×10^{-9}	8.8×10^{-11}
Sodium fire	2.4×10^{-8}	2.8×10^{-6}	1.7×10^{-10}
Design-basis earthquake	1.9×10^{-6}	0.00022	4.9×10^{-8}
Waste handling spill	1.8×10^{-8}	2.2×10^{-6}	4.8×10^{-10}

[a] Increased likelihood of a latent cancer fatality.
[b] Increased number of latent cancer fatalities.

Table 4–44 Nonradiological Impacts of Accidents Under Alternative 4

Accident	*Frequency (Event Per Year)*	*Receptor Location*	*Exposure*
Uranium handling accident	0.01	Noninvolved worker	Uranium: less than ERPG-1
		Maximally exposed offsite individual	Uranium: less than ERPG-1
Design-basis earthquake	0.0002	Noninvolved worker	Uranium: less than ERPG-1
		Maximally exposed offsite individual	Uranium: less than ERPG-1
Sodium fire	0.00001	Noninvolved worker	Sodium: less than ERPG-1
		Maximally exposed offsite individual	Sodium: less than ERPG-1
Uranium fire	0.00001	Noninvolved worker	Uranium: less than ERPG-1
		Maximally exposed offsite individual	Uranium: less than ERPG-1
Beyond-design-basis earthquake	0.00001	Noninvolved worker	Cadmium: less than ERPG-1
			Uranium: less than ERPG-1
		Maximally exposed offsite individual	Cadmium: less than ERPG-1
			Uranium: less than ERPG-1

ERPG = Emergency Response Planning Guideline.

4.6.5 Environmental Justice

As discussed in Appendix H, Executive Order 12898 directs Federal agencies to address the disproportionately high and adverse health or environmental effects of alternatives on minority or low-income populations.

Analyses of normal operations and accident conditions presented in Section 4.6.4 show the risk of latent cancer fatalities to the public residing within 80 kilometers (50 miles) of the electrometallurgical and melt and dilute treatment processing facilities at ANL-W to be much lower than 1. Therefore, radiological and nonradiological risks posed by implementation of this alternative would not result in disproportionately high and adverse consequences to any particular group within the general population, including minority or low-income populations.

4.6.6 Waste Management

This alternative would generate process waste from treatment operations, other associated process waste from normal support operations, and deactivation waste following the conclusion of operations. Process waste would include high-level radioactive metallic and ceramic waste. Other associated process waste would include operational waste such as failed equipment, rags, packaging materials, and other miscellaneous items. Deactivation waste would include the disposal of process equipment and other materials. The fuel hardware in this alternative is used as additional steel in the melt and dilute process. All of these materials would be categorized according to existing DOE Orders and ANL-W waste management procedures. The anticipated categorization of the waste types generated and their expected interim storage and final disposal locations are given in Table 4–9 (see Section 4.2.6). The quantities of various waste forms generated as a result of Alternative 4 are provided in **Table 4–45**.

Estimates of the total amount of other associated process waste that would be generated are based on an evaluation of waste forecasts from ANL-W that accounted only for the fraction of total ANL-W waste that would be attributable to the treatment of sodium-bonded spent nuclear fuel. The values in Table 4–45 are for disposal and account for volume reduction. It is anticipated that a large fraction of the low-level radioactive waste that would be generated as a result of Alternative 4 could be volume-reduced at the Waste

Experimental Reduction Facility at INEEL prior to disposal at the INEEL Subsurface Disposal Area at the Radioactive Waste Management Complex.

Table 4–45 Amounts of Waste Generated at ANL-W Under Alternative 4[a]

Waste Stream	*Waste Quantities*	
	Volume (cubic meters)	*Mass (kilograms)*
Direct Process Waste		
Fuel assembly hardware (low-level radioactive waste)	0	0
High-level radioactive ceramic waste	16.3 (26 canisters) [b]	24,400
High-level radioactive metallic waste	1.3 (2 canisters) [b]	2,500
Melt and dilute product	45.6 (114 canisters) [b]	114,000
Other Associated Process Waste		
High-level radioactive waste	0.4 (1 canister) [b]	220
Low-level radioactive waste [c]	650	132,000
Transuranic waste	11.2	4,730
Mixed waste	32.1	17,300
Sanitary waste	4,960	1.72×10^{6}
Deactivation Waste		
Low-level radioactive waste [c]	195	66,000
Transuranic waste	1.6	853
Mixed waste	5.6	3,600

[a] These waste generation estimates are through the year 2015. This is the assumed date that these materials might be sent to the repository. Treatment, high-level radioactive waste processing, deactivation, and interim storage would be accomplished during this time period.
[b] Standardized canisters.
[c] The volumes listed represent final disposal volumes following volume reduction at the Waste Experimental Reduction Facility at INEEL.
Source: ANL 1999.

The waste values in Table 4–45 are the total quantities that would be produced as a result of Alternative 4 operations. They are not incremental increases over the volumes provided in Table 4–10 that would result from the No Action Alternative. In Alternative 4, both the driver and blanket spent nuclear fuel would be transformed into high-level radioactive waste forms (ceramic and metallic, and a melt and dilute product) for disposal in the repository. In this conversion process, the total volume of material to be disposed of in the repository would be reduced from direct disposal values.

Direct Process Waste

For this alternative, fuel assembly hardware would be used as part of the required stainless steel to form the material ingot for disposal of the blanket spent nuclear fuel by melting. Its mass is included as part of the spent nuclear fuel disposal.

Under this alternative, metallic and ceramic high-level radioactive waste would be a primary product of the electrometallurgical treatment of driver spent nuclear fuel. The salt removed from the electrorefiners would contain the majority of fission products and transuranics from the spent nuclear fuel. This removed salt would be packaged and transferred to the Hot Fuel Examination Facility for processing into ceramic waste. The metallic waste form would consist primarily of stainless steel cladding hulls containing the noble metal fission products. The hulls would be removed from the electrorefiner and packaged for shipment to the Hot Fuel Examination Facility for processing into the metallic waste form. Both the ceramic and metallic waste

would be categorized as high-level radioactive waste. The volumes of waste forms provided in Table 4–45 are for the standardized canisters required for disposal of these materials.

The metallic and ceramic high-level radioactive waste and the melt and dilute product that would be generated at ANL-W would be stored temporarily at the Radioactive Scrap and Waste Facility at ANL-W to allow retrieval for future disposal. The Radioactive Scrap and Waste Facility was designed and constructed for temporary storage of this type of waste. Shielding would be provided by a combination of (1) steel storage liners which would store the waste, and (2) the soil surrounding the liners. When a geologic repository is available, the waste cans containing these materials would be removed from storage, shipped to the INEEL Dry Transfer Facility, and prepared for shipment to the repository.

Other Associated Process High-Level Radioactive Waste

High-level radioactive waste would be generated as a result of blanket spent nuclear fuel processing at ANL-W. This would result from activities in the Hot Fuel Examination Facility. Material in this waste stream would consist of the absorbent used in the off-gas system which has collected the volatile radionuclides released from the spent nuclear fuel when heated.

The volume of high-level radioactive waste that would be generated is expected to be less than the amount needed to fill a single standardized waste canister. Conservatively, the volume of a single canister, 0.4 cubic meters, has been used for the volume of high-level radioactive waste generated.

Other Associated Process Low-Level Radioactive Waste

Low-level radioactive waste would be generated as a result of processing sodium-bonded spent nuclear fuel at ANL-W. This would result from activities in the Fuel Conditioning Facility and the Hot Fuel Examination Facility (e.g., equipment decontamination and repair), as well as in other facilities at ANL-W (e.g., analytical laboratory activities). Material in this waste stream has been generated and routinely handled at ANL-W for many years.

The volume of low-level radioactive waste resulting from electrometallurgical and melt and dilute treatment processing of sodium-bonded spent nuclear fuel at ANL-W that would require disposal (after volume reduction) would be approximately 50 cubic meters (1,766 cubic feet) per year. This represents approximately 1 percent of the total annual volume of low-level radioactive waste currently being disposed of at the INEEL Subsurface Disposal Area at the Radioactive Waste Management Complex, and the total 650 cubic meters (22,955 cubic feet) represent approximately 0.6 percent of the total Radioactive Waste Management Complex disposal capacity.

Other Associated Process Transuranic Waste

Transuranic waste would be generated from decontamination activities for repair and maintenance of items, and miscellaneous work associated with processing the sodium-bonded spent nuclear fuel. Transuranic waste would be generated primarily from activities conducted in gloveboxes and hot cells at ANL-W.

All of the transuranic waste that would be generated at ANL-W would be packaged and certified in accordance with Waste Isolation Pilot Plant Acceptance Criteria prior to transport to the Waste Isolation Pilot Plant. The transuranic waste generated would amount to approximately 1 cubic meter (35 cubic feet) per year, which would be approximately 0.002 percent of the volume of transuranic waste in retrievable storage at the Radioactive Waste Management Complex at INEEL. The total volume of transuranic waste would be 11.2 cubic meters (395 cubic feet), which would be less than 0.006 percent of the estimated total volume of transuranic waste to be placed at the Waste Isolation Pilot Plant.

Other Associated Process Mixed Waste

Mixed waste would be generated primarily from the disposal of any cadmium-contaminated equipment or cleanup material and the analysis of cadmium samples. Mixed waste would be handled according to ANL-W procedures that require limited accumulation at the point of generation. Interim storage of this waste would be accomplished at the Radioactive Scrap and Waste Facility prior to eventual disposal. The Radioactive Scrap and Waste Facility is a permitted mixed waste storage facility for these materials. The mixed waste streams that contribute to the overall mixed waste generated by electrometallurgical treatment have been identified in the INEEL Site Treatment Plan (DOE 1995b).

Deactivation Waste

A variety of waste would be generated as part of deactivation activities associated with processing at ANL-W. This waste would include process equipment and process material, such as electrorefiner cadmium from electrometallurgical treatment of driver spent nuclear fuel. Generated waste categories would include low-level radioactive waste, transuranic waste, and mixed waste. This waste would be categorized and disposed of according to DOE Orders and ANL-W radioactive waste management procedures, as described above for each waste category.

The largest volume of deactivation waste would be low-level radioactive waste, generated as a result of dismantling and disposal (electrometallurgical treatment and melt and dilute equipment). Components of electrometallurgical treatment of the driver spent nuclear fuel that would require disposition include two electrorefiners, two hot hydrostatic presses, and two V-mixers, as well as other components such as the grinder/crusher. Decontamination of these components would generate additional mixed, transuranic, and low-level radioactive waste that would require management. The deactivation waste volume would be generated in a single year. This waste would represent an increase of approximately three times the annual waste generated by the treatment operations of Alternative 4. The total deactivation waste would represent an additional 30 percent over the total associated process waste requiring disposal.

4.7 ALTERNATIVE 5: DECLAD AND CLEAN BLANKET FUEL AND ELECTROMETALLURGICALLY TREAT DRIVER FUEL AT ANL-W; MELT AND DILUTE BLANKET FUEL AT SRS

Under this alternative, the sodium-bonded blanket spent nuclear fuel would be declad and cleaned to remove metallic sodium at ANL-W, packaged in aluminum cans, and shipped to SRS for treatment using the melt and dilute process at Building 105-L. The melt and dilute product from the treatment process would be stored at SRS pending repackaging and transportation for disposal in a geologic repository. The removed sodium would be stabilized using an oxidation/carbonation process (ANL 1999). The sodium-bonded driver spent nuclear fuel would be treated at ANL-W using the electrometallurgical treatment process. The process steps for the electrometallurgical treatment of driver spent nuclear fuel would be similar to those described earlier in Section 4.3 and in Appendix C. The treatment of driver spent nuclear fuel using the electrometallurgical process could start as early as 2000 and could be completed by 2006 to 2007. The preparation of blanket spent nuclear fuel and its shipment to SRS could start in 2003 and could be completed by 2009. In addition, a full year of operations would be needed to deactivate the processing equipment and establish an industrially safe configuration.

Current planning at SRS has scheduled the melt and dilute process at Building 105-L for other missions until 2035 (DOE 2000). Melt and dilute process of blanket spent nuclear fuel at SRS could start around 2020, if capacity becomes available, and could be completed by 2023.

4.7.1 Air Quality

Nonradiological Gaseous Emissions

It is expected that criteria and hazardous air pollutants released from operational activities at ANL-W under this alternative would be the same as for Alternative 1, as described in Section 4.3.1 (see also Appendix E, Section E.5.3.1, for more detail). Baseline air quality concentrations are presented in Section 3.2.3.1.

At SRS, nonradiological air emissions would result from operation of ancillary support facilities for the melt and dilute process at Building 105-L. The concentrations of nonradiological air pollutants attributed to this alternative at SRS are presented in **Table 4–46**. These concentrations are based on information in the SRS Spent Nuclear Fuel Management Final EIS (DOE 2000) for the melt and dilute processing of similar fuel (see Appendix E, Section E.5.3.2, for more details). The site boundary concentrations are the incremental concentrations that would be generated in this alternative plus the baseline concentrations given in Section 3.3.3.1. Only those air pollutants that are expected to be emitted under this alternative and have ambient air quality standards are presented in the table. Note that SRS has no Prevention of Significant Deterioration increment-consuming sources on site; therefore, a Prevention of Significant Deterioration increment analysis was not performed. SRS is located in an area of attainment for criteria pollutants; therefore, a confirmatory analysis is not required for this alternative. Health effects from hazardous chemicals associated with this alternative are addressed in Section 4.7.4.1.

Radiological Gaseous Emissions

The decladding and cleaning of the blanket spent nuclear fuel and the electrometallurgical treatment of the driver spent nuclear fuel at ANL-W would release gaseous fission products to the argon cell environment. Krypton-85 and elemental tritium would be the most prevalent radioactive gaseous fission products released to the environment. The tritium released in the cell would not be oxidized due to a very low presence of oxygen and humidity in the argon cell. The argon cell also would contain an equilibrium concentration of other radionuclide isotopes. Appendix E, Section E.4.1, provides a list of various isotopes that would be present in the argon cell in nanocuries (10^{-9} curies) and would be released to the atmosphere through the facility stack along with krypton-85 and elemental tritium. The maximum release of radioactive gaseous emissions would occur when decladding the blanket spent nuclear fuel for packaging and shipment to SRS and electrometallurgical treatment of driver spent nuclear fuel are performed simultaneously. It was estimated that this simultaneous operation would occur over a three-year period starting in 2003 (see Appendix E, Section E.4.2, for details on releases during the processing period at ANL-W under this alternative). Based on an annual decladding throughput of 10 metric tons of heavy metal of blanket spent nuclear fuel and an electrometallurgical treatment process of about 0.6 metric tons of heavy metal of driver spent nuclear fuel, about 809 curies of elemental tritium and 11,860 curies of gaseous krypton-85 would be released annually to the atmosphere.

Since declad and cleaned fuel would be packaged and sent to SRS, some gaseous fission products would be expected in that fuel. However, it was conservatively assumed that the gaseous fission products in the blanket spent nuclear fuel also would be released to the environment during the melt and dilute process at SRS. The radiological exposures of the public and workers from radioactive emissions are presented in Section 4.7.4.

4.7.2 Water Resources

As stated in Section 4.4.2, the decladding and cleaning of blanket spent nuclear fuel and electrometallurgical treatment of driver spent nuclear fuel would not discharge any radiological chemical material to the surface or groundwater at the INEEL site. These activities also would not impact the current groundwater usage at the site. For a discussion of impacts on water resources at ANL-W; see Section 4.4.2.

Table 4–46 Nonradiological Air Quality Concentrations at the Site Boundary Under Alternative 5 at SRS for Comparison With Ambient Air Quality Standards

Pollutant	*Averaging Period*	*Most Stringent Standard or Guideline (micrograms per cubic meter)*	*Maximum Incremental Concentration (micrograms per cubic meter)*
Criteria Pollutants			
Carbon monoxide	8 hours 1 hour	10,000 40,000	0.08 0.51
Nitrogen dioxide	Annual	100	Less than 0.01
PM_{10}	Annual 24 hours (interim) 24 hours (99^{th} percentile over 3 years)	50 150 150	Not detectable Not detectable Not available
$PM_{2.5}$	3-year annual 24 hours (98^{th} percentile over 3 years)	15 65	Not available Not available
Sulfur dioxide	Annual 24 hours 3 hours	80 365 1,300	0.01 0.03 Not detectable
State-regulated Pollutants			
Gaseous fluoride	30 days 7 days 24 hours 12 hours	0.8 1.6 2.9 3.7	Not detectable Not detectable Not detectable Not detectable
Total suspended particulates	Annual	75	Less than 0.01
Hazardous/Toxic Compounds			
1,1,1-trichloroethane	24 hours	9,550	Less than 0.01
Benzene	24 hours	150	Not detectable
Ethanolamine	24 hours	200	Less than 0.01
Ethyl benzene	24 hours	4,350	Not detectable
Ethylene glycol	24 hours	650	Less than 0.01
Formaldehyde	24 hours	15	Less than 0.01
Glycol ethers	24 hours	No standard	Less than 0.01
Hexachloronaphthalene	24 hours	1	Less than 0.01
Hexane	24 hours	900	Less than 0.01
Manganese	24 hours	25	Not detectable
Methyl alcohol	24 hours	1,310	Less than 0.01
Methyl-ethyl-ketone	24 hours	14,750	Less than 0.01
Methyl-isobutyl-ketone	24 hours	2,050	Not detectable
Methylene chloride	24 hours	8,750	Not detectable
Naphthalene	24 hours	1,250	Less than 0.01
Nitric acid	24 hours	125	Not detectable
Phenol	24 hours	190	Not detectable
Phosphorous	24 hours	0.5	Not detectable
Sodium hydroxide	24 hours	50	Not detectable
Toluene	24 hours	2,000	Less than 0.01
Trichloroethane	24 hours	6,750	Not detectable
Vinyl acetate	24 hours	176	Not detectable
Xylene	24 hours	4,350	Less than 0.01

PM_n = Particulate matter less than or equal to *n* microns in diameter.
Source: Bickford et al. 1997.

The impacts on water resources from treating blanket spent nuclear fuel at Building 105-L using the melt and dilute process are described below.

Surface Water

No surface water would be used for the melt and dilute processing of blanket spent nuclear fuel at Building 105-L. Building 105-L is outside the 100-year floodplain, as shown in Figure 3-6.

Nonradiological Liquid Effluent

No nonradiological liquid effluent would be generated by melting and diluting blanket spent nuclear fuel at Building 105-L. Sanitary waste would be treated at the SRS Central Wastewater Treatment Facility and discharged through an existing NPDES-permitted outfall. Since employment would not increase as a result of processing this fuel, the treatment rates through the Central Wastewater Treatment Facility would not be affected and the requirements of the SRS NPDES permit would continue to be met (DOE 2000).

Radiological Liquid Effluent

No radiological liquid effluent would be discharged to surface water from the melt and dilute process at Building 105-L.

Groundwater

Process water would not be required for the melt and dilute process at Building 105-L. Domestic water would come from groundwater. No increase in domestic water use is anticipated since no increase in employment is expected to result from the melt and dilute operation.

Nonradiological Liquid Effluent

No nonradiological chemicals would be discharged to groundwater from the melt and dilute processing at Building 105-L.

Radiological Liquid Effluent

No radiological liquid effluent would be discharged to groundwater from the melt and dilute process at Building 105-L.

4.7.3 Socioeconomics

Under this alternative, the existing facilities at ANL-W and SRS would remain operational. No new employment or in-migration of workers would be required. Thus, there would be no additional impacts on the socioeconomic conditions in the region around INEEL and SRS.

4.7.4 Public and Occupational Health and Safety

The assessments of potential incremental radiological and chemical impacts associated with this alternative are presented in this section. Summaries of radiological and chemical impacts from normal operations are presented in Tables 4–47 through 4–48 for the public and workers, respectively. The radiological impacts from a spectrum of hypothetical accident scenarios are provided in Tables 4–50 through 4–53. The impacts from hazardous chemical releases during accident conditions are similar to those presented in Section 4.5.4.1.

Background information on the effects of radiation on human health and safety is presented in Appendix E, Section E.2.

4.7.4.1 Normal Operations

Radiological Impacts

Under this alternative, radioactive releases would occur during sodium-bonded blanket spent nuclear fuel decladding and cleaning, driver spent nuclear fuel chopping, and electrorefining. All of these activities would be performed in the argon cell. Appendix E, Sections E.3, E.4.1, and E.4.2, details the method and assumptions used for calculating the impacts of normal operational radiological releases on the public health and safety. The maximum annual dose to the public at ANL-W would result when decladding and cleaning of blanket spent nuclear fuel and treatment of driver spent nuclear fuel are performed simultaneously under this alternative. The doses from decladding and cleaning blanket spent nuclear fuel and treating driver spent nuclear fuel at ANL-W would be similar to those presented for Alternative 2 in Section 4.4.4.1.

Calculated maximum annual and projected total radiological impacts to the public are given in **Table 4–47**. The impacts are calculated for two types of receptors: the general public living within 80 kilometers (50 miles) of ANL-W and Building 105-L at SRS in the year 2010, and a maximally exposed offsite individual (a member of the public assumed to be residing at the INEEL or SRS site boundary and receiving the maximum dose). Primary contributors to doses to members of the public at ANL-W would be from releases of tritium gas (about 1 percent of which were assumed conservatively to be in oxidized form) and krypton-85, which together would contribute over 99.9 percent of the total calculated doses. As shown in this table, the expected radiation doses to the maximally exposed offsite individual and the general public would be much smaller than the limit of 10 millirem per year set by the EPA (40 CFR 61) and DOE (DOE Order 5400.5).

The blanket spent nuclear fuel would be declad and cleaned at ANL-W, where it is expected that the gaseous fission products would be released. However, for the melt and dilute process, it is assumed conservatively that these gaseous fission products would be released at SRS. The melt and dilute process is assumed to continue for three years. (Appendix E, Section E.4.4, provides the details on the treatment process duration.) To put the operational impacts into perspective, comparisons with impacts from natural background radiation also are included in the table.

Table 4–48 summarizes worker population doses. Occupational doses were estimated by examining the type and duration of various operations performed by SRS workers involved with the melt and dilute process. The estimated annual collective worker dose would be 50 person-rem, with an average individual dose of 500 millirem per year for each of the 100 involved workers. If these estimates were projected for maximum process activities over three years, the project total worker population dose would be 150 person-rem, leading to a risk of 0.06 latent cancer fatalities. The estimated annual collective worker dose from decladding and cleaning blanket spent nuclear fuel and electrometallurgically treating driver spent nuclear fuel at ANL-W is 22 person rem, as indicated in Section 4.4.4.1.

As shown in Tables 4–47 and 4–48:

- The annual dose to the maximally exposed offsite individual at ANL-W would be 0.00038 millirem per year, with an associated risk of developing a fatal cancer of 1.9×10^{-10} per year (or one chance in 5.3 billion that the individual would develop a fatal cancer per year of exposure).

Table 4–47 Annual and Project Total Radiological Impacts to the Public From Operational Activities Under Alternative 5

Receptor	*Melt and Dilute Blanket Spent Nuclear Fuel at SRS*	*Clean Blanket Spent Nuclear Fuel and Electrometallurgically Treat Driver Spent Nuclear Fuel at ANL-W*
Population Within 80 Kilometers (50 Miles) in the Year 2010		
Collective dose (person-rem per year) [a]	0.0076	0.0030
Excess latent cancer fatalities (per year)	3.8×10^{-6}	1.5×10^{-6}
Project total excess latent cancer fatalities [b]	0.000011	8.3×10^{-6}
Maximally Exposed Offsite Individual		
Dose (millirem per year) [a]	0.00010	0.00038
Percent of annual background radiation [c]	0.000033	0.00011
Latent cancer fatality risk (per year)	5.0×10^{-11}	1.9×10^{-10}
Project total lifetime cancer fatality risk	1.5×10^{-10}	1.0×10^{-9}
Average Individual Within 80 Kilometers (50 Miles)		
Dose (millirem per year) [d]	0.000011	0.000012
Latent cancer fatality risk (per year)	5.5×10^{-12}	6.2×10^{-12}
Project total lifetime cancer fatality risk [b]	1.6×10^{-11}	3.5×10^{-11}

[a] Annual maximum dose during normal operations.
[b] Total calculated risk over nine years at ANL-W and three years at SRS.
[c] The annual natural background radiation level at INEEL and at SRS is about 360 and 300 millirem, respectively, for the average individual (see Tables 3–8 and 3–20); the population within 80 kilometers (50 miles) in the year 2010 would receive 86,500 person-rem at INEEL and 254,000 person-rem at SRS.
[d] Obtained by dividing the population dose by the number of people projected to live in the year 2010 within 80 kilometers (50 miles) of ANL-W (240,338) and SRS Building 105-L (715,000).

Table 4–48 Annual and Project Total Radiological Impacts to Workers From Operational Activities Under Alternative 5

Receptor	*Impacts*	
Worker [a]	Operations at SRS	Operations at ANL-W
Average worker dose (millirem per year)	500	60
Average worker latent cancer fatality risk (project total)	0.00060 [b]	0.00022 [c]
Worker Population		
Collective dose (person-rem per year)	50	22
Excess latent cancer fatalities (per year)	0.020	0.0088
Project total dose (person-rem)	150	231
Project total excess latent cancer fatalities	0.06 [b]	0.092

[a] The regulatory dose limit for an individual worker is 5,000 millirem per year (10 CFR 835). However, the maximum annual dose to a worker would be kept below the DOE Administrative Control Level of 2,000 millirem per year, as established for all DOE activities in DOE Order N 441.1.
[b] Operations at SRS to treat blanket spent nuclear fuel using melt and dilute processing at Building 105-L would be performed over three years.
[c] Operations at ANL-W to declad and clean blanket spent nuclear fuel and treat driver spent nuclear fuel would be performed over nine years. The project total dose includes 33 person-rem from one year of facility deactivation activities.

Sources: ANL 1999, DOE 2000.

- The collective dose to the population within 80 kilometers (50 miles) of the ANL-W facilities would be 0.0030 person-rem per year, with an associated 1.5×10^{-6} latent cancer fatalities per year (or one chance in 670,000 that the population would experience a fatal cancer per year of exposure).

- The project total risk to the population within 80 kilometers (50 miles) of the ANL-W facilities would be 8.3×10^{-6} latent cancer fatalities (or one chance in 120,000 that the exposed population would experience a fatal cancer).

- The collective dose to ANL-W facility workers would be 22 person-rem per year, with an associated 0.0088 latent cancer fatalities per year (or one chance in 113 that the workers would experience a fatal cancer per year of operation).

- The project total dose to ANL-W facility workers would be 231 person-rem, with an associated 0.092 latent cancer fatalities (or one chance in 11 that the exposed workers would experience a fatal cancer).

- The annual dose to the maximally exposed offsite individual from melt and dilute processing at Building 105-L would be 0.00010 millirem per year, with an associated risk of developing a fatal cancer of 5×10^{-11} per year (or one chance in 20 billion that the individual would develop a fatal cancer per year of exposure).

- The collective dose to the population within 80 kilometers (50 miles) of Building 105-L would be 0.0076 person-rem per year, with an associated 3.8×10^{-6} latent cancer fatalities per year (or one chance in 263,100 that the population would experience a fatal cancer per year of exposure).

- The project total risk to the population within 80 kilometers (50 miles) of Building 105-L would be 0.000011 latent cancer fatalities (or one chance in 91,000 that the exposed population would experience a fatal cancer).

- The collective dose to Building 105-L facility workers would be 50 person-rem per year, with an associated 0.020 latent cancer fatalities (or one chance in 50 that the workers would experience a fatal cancer per year of operation).

- The project total dose to Building 105-L facility workers would be 150 person-rem, with an associated 0.06 latent cancer fatalities (or one chance in 17 that the exposed workers would experience a fatal cancer).

These results indicate that no increase in the expected number of fatal cancers among the various affected populations would result from the activities under this alternative.

Hazardous Chemical Impacts

The hazardous chemical impacts to the public residing at the INEEL site boundary from the operational activities at ANL-W under Alternative 5 would be similar to the impacts evaluated for Alternative 1, as described in Section 4.3.4.1. The results indicate that no adverse toxic health or cancer effects would be expected from exposure to hazardous chemicals released under this alternative. The existing baseline chemical environment is presented in Section 3.2.10.2.

For SRS, both carcinogenic and noncarcinogenic health effects to the public were assessed from exposure to hazardous chemicals, and the results are summarized in **Table 4–49**. The results indicate that no adverse toxic health or cancer effects would be expected from exposure to hazardous chemicals released at SRS

under this alternative (see Appendix E, Section E.5.3, for more details). The existing baseline chemical environment is presented in Section 3.3.10.2.

Table 4–49 Hazardous Chemical Impacts to the Public From Operational Activities at SRS Under Alternative 5

Chemical	*Annual Concentration (milligrams per cubic meter)*	*Hazard Quotient (noncarcinogenic chemicals)*	*Cancer Risk (carcinogenic chemicals)*
Formaldehyde	1.3×10^{-6}	None	1.6×10^{-8}
Hexane	1.3×10^{-6}	6.3×10^{-6}	None
Manganese	Not detectable	Not detectable	None
Methyl ethyl ketone	1.3×10^{-6}	1.3×10^{-6}	None
Naphthalene	1.3×10^{-6}	0.00042	None
Toluene	1.3×10^{-6}	3.1×10^{-6}	None
Hazard Index		0.00043	Not applicable

4.7.4.2 Facility Accidents

Radiological Impacts

The potential radiological impacts to the public and a noninvolved onsite worker resulting from accidents during decladding and cleaning and melting and diluting the blanket spent nuclear fuel elements, and from electrometallurgical treatment of driver spent nuclear fuel operational activities at ANL-W and SRS, are summarized and presented in this section. The detailed analysis of facility accidents, with associated assumptions, is presented in Appendix F. The detailed analysis considered a wide spectrum of potential accident scenarios, including fire, spills, criticality, earthquake, and aircraft crash. The aircraft crash and criticality events were determined to have an occurrence frequency of less than 10^{-7} per year, and consequence analyses for these two events were not performed. Processing of the blanket spent nuclear fuel would be performed in the Hot Fuel Examination Facility; treatment of the driver spent nuclear fuel is performed in both the Hot Fuel Examination Facility and the Fuel Conditioning Facility. Because the processing of the driver spent nuclear fuel would take place in both of these facilities, the beyond-design-basis earthquake event is assessed for the driver spent nuclear fuel, taking into account the multi-facility impacts of this event. The decladding and cleaning of the blanket spent nuclear fuel would be performed only in the Hot Fuel Examination Facility. The multi-facility impacts of the beyond-design-basis earthquake are not relevant to this blanket spent nuclear fuel processing. Therefore, only the higher frequency design-basis earthquake event was analyzed. **Table 4–50** presents the frequencies and consequences of the postulated set of accidents to the maximally exposed offsite individual; the offsite population residing within 80 kilometers (50 miles) of the facility; and a noninvolved worker located 100 meters (330 feet) to 230 meters (755 foot) from the facility. The 230-meter (755-foot) distance is the ANL-W bus staging area, which leads to a higher dose to the noninvolved worker for the scenarios with elevated releases.

The dose to the maximally exposed offsite individual was calculated for the 95th percentile meteorological conditions. The doses to the population and the noninvolved worker were calculated using 50th percentile meteorological conditions. DOE did not quantitatively estimate the involved worker dose due to accidents (see the discussion on the involved worker in Section 4.2.4.2). The accident risks for the same receptors are summarized in **Table 4–51**.

Table 4–50 Accident Frequency and Consequences at ANL-W Under Alternative 5

Accident	Frequency (event per year)	Maximally Exposed Offsite Individual		Population Within 80 Kilometers (50 Miles)		Noninvolved Worker	
		Dose (millirem)	Latent Cancer Fatality Risk [a]	Dose (person-rem)	Excess Latent Cancer Fatalities [b]	Dose (millirem)	Latent Cancer Fatality Risk [a]
Driver Spent Nuclear Fuel							
Salt powder spill	0.01	0.00046	2.3×10^{-10}	0.000098	4.9×10^{-8}	4.7×10^{-7}	1.9×10^{-13}
Salt transfer drop	1.0×10^{-7}	0.19	9.5×10^{-8}	0.022	0.000011	0.073	2.9×10^{-8}
Transuranic waste fire	0.001	0.059	3.0×10^{-8}	0.0071	3.6×10^{-6}	0.22	8.8×10^{-8}
Cask drop	0.01	0.030	1.5×10^{-8}	0.0035	1.8×10^{-6}	0.00084	3.4×10^{-10}
Design-basis earthquake	0.008	12	6.0×10^{-6}	1.4	0.0007	4.7	1.9×10^{-6}
Beyond-design-basis earthquake	0.00001	22,000	0.022	2,500	1.3	370	0.00015
Blanket Spent Nuclear Fuel							
Cask drop	0.01	0.0024	1.2×10^{-9}	0.00028	1.4×10^{-7}	0.000049	2.0×10^{-11}
Transuranic waste fire	0.001	0.059	3.0×10^{-8}	0.0071	3.6×10^{-6}	0.22	8.8×10^{-8}
Sodium fire [c]	0.008	5.9	3.0×10^{-6}	0.69	0.00035	0.054	2.2×10^{-8}

[a] Increased likelihood of a latent cancer fatality.
[b] Increased number of latent cancer fatalities.
[c] The frequency for this accident is the frequency for the facility design-basis earthquake-initiating cell fire.

Table 4–51 Annual Cancer Risks Due to Accidents at ANL-W Under Alternative 5

Accident	Maximally Exposed Offsite Individual [a]	Population Within 80 Kilometers (50 Miles) [b]	Noninvolved Worker [a]
Driver Spent Nuclear Fuel			
Salt powder spill	2.3×10^{-12}	4.9×10^{-10}	1.9×10^{-15}
Salt transfer drop	9.5×10^{-15}	1.1×10^{-12}	2.9×10^{-15}
Transuranic waste fire	3.0×10^{-11}	3.6×10^{-9}	8.8×10^{-11}
Cask drop	1.5×10^{-10}	1.7×10^{-8}	3.4×10^{-12}
Design-basis earthquake	4.8×10^{-8}	5.6×10^{-6}	1.5×10^{-8}
Beyond-design-basis earthquake	2.2×10^{-7}	0.000013	1.5×10^{-9}
Blanket Spent Nuclear Fuel			
Cask drop	1.2×10^{-11}	1.4×10^{-9}	2.0×10^{-13}
Transuranic waste fire	3.0×10^{-11}	3.6×10^{-9}	8.8×10^{-11}
Sodium fire	2.4×10^{-8}	2.8×10^{-6}	1.7×10^{-10}

[a] Increased likelihood of a latent cancer fatality.
[b] Increased number of latent cancer fatalities.

For accidents at ANL-W, the highest risk of a latent cancer fatality to the maximally exposed offsite individual and to a noninvolved worker would be 2.2×10^{-7} per year (or one chance in 4.5 million that the individual would develop a fatal cancer per year of operation) and 1.5×10^{-8} per year (or one chance in 66.7 million that the individual would develop a fatal cancer per year of operation), respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.000013 per year (or one chance in 76,920 that the population would experience a fatal cancer per year of operation).

The potential radiological impacts to the public and a noninvolved onsite worker due to accidents during melt and dilute operational activities at SRS are summarized below. The detailed analysis of facility accidents, with the associated assumptions, is presented in Appendix F. **Table 4–52** presents the frequencies and consequences of the postulated set of accidents to the maximally exposed offsite individual, the offsite population residing within 80 kilometers (50 miles) of the facility, and a noninvolved worker located 100 meters (330 feet) to 300 meters (980 feet) from the facility. The 300-meter (980-foot) distance leads to a higher dose to the noninvolved worker for the scenarios with elevated releases.

Table 4–52 Accident Frequency and Consequences at SRS Under Alternative 5

Accident	*Frequency (event per year)*	*Maximally Exposed Offsite Individual*		*Population Within 80 Kilometers (50 Miles)*		*Noninvolved Worker*	
		Dose (millirem)	*Latent Cancer Fatality Risk* [a]	*Dose (person-rem)*	*Excess Latent Cancer Fatalities* [b]	*Dose (millirem)*	*Latent Cancer Fatality Risk* [a]
Waste handling spill	0.0064	2.1	1.1×10^{-6}	3.6	0.0018	0.17	6.8×10^{-8}
Loss of power	0.006	2100	0.0011	3500	1.8	140	0.000056
Melter eruption/explosion [c]	0.0005	269	0.00014	1160	0.58	72.9	0.000029
Fire	0.075	86	0.000043	140	0.07	6.3	2.5×10^{-6}

[a] Increased likelihood of a latent cancer fatality.
[b] Increased number of latent cancer fatalities.
[c] In the draft EIS, this accident was identified as "loss of cooling water." Consistent with the SRS Spent Nuclear Fuel Management Final EIS, the accident name was changed.

The dose to the maximally exposed offsite individual was calculated for the 95th percentile meteorological conditions. The doses to the population and the noninvolved worker were calculated using 50th percentile meteorological conditions. DOE did not quantitatively estimate the facility worker population dose due to accidents. The accident risks for the same receptors are summarized in **Table 4–53**.

Table 4–53 Annual Cancer Risks Due to Accidents at SRS Under Alternative 5

Accident	*Maximally Exposed Offsite Individual* [a]	*Population Within 80 Kilometers (50 Miles)* [b]	*Noninvolved Worker* [a]
Waste handling spill	6.7×10^{-9}	0.000012	5.5×10^{-10}
Loss of power	6.6×10^{-6}	0.011	3.4×10^{-7}
Melter eruption/explosion	7.0×10^{-8}	0.00029	1.5×10^{-8}
Fire	3.2×10^{-6}	0.0053	1.9×10^{-7}

[a] Increased likelihood of a latent cancer fatality.
[b] Increased number of latent cancer fatalities.

For accidents at SRS, the highest risk of a latent cancer fatality to the maximally exposed offsite individual and to a noninvolved worker would be 6.6×10^{-6} per year (or one chance in 151,500 that the individual would develop a fatal cancer per year of operation) and 3.4×10^{-7} per year (or one chance in 2.9 million that the individual would develop a fatal cancer per year of operation), respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.011 per year (or one chance in 91 that the population would experience a fatal cancer per year of operation).

Hazardous Chemical Impacts

The impacts of accidents involving hazardous chemicals for this alternative are the same as those described in Section 4.5.4.2 for Alternative 3: Declad and clean blanket fuel and electrometallurgically treat driver fuel at ANL-W; PUREX process blanket fuel at SRS.

4.7.5 Environmental Justice

As discussed in Appendix H, Executive Order 12898 directs Federal agencies to address disproportionately high and adverse health or environmental effects of alternatives on minority or low-income populations.

Analyses of normal operations and accident conditions presented in Section 4.7.4 show the risk of latent cancer fatalities to the public residing within 80 kilometers (50 miles) of the electrometallurgical treatment and decladding and cleaning processing facilities at ANL-W and of the melt and dilute treatment facility at SRS to be much lower than 1. Therefore, radiological and nonradiological risks posed by implementation of this alternative would have no disproportionately high and adverse consequences on any particular group within the general population, including minority or low-income populations.

4.7.6 Waste Management

ANL-W

This alternative would generate process waste from treatment operations, other associated process waste from normal support operations, and deactivation waste following the conclusion of operations. Process waste would include fuel hardware and high-level radioactive metallic and ceramic waste. Other associated process waste would include operational waste such as failed equipment, rags, packaging materials, and other miscellaneous items. Deactivation waste would include the disposal of process equipment and other materials. All of these materials would be categorized according to existing DOE Orders and ANL-W waste management procedures. The anticipated categorization of these waste types generated at ANL-W and their expected interim storage and final disposal locations are given in Table 4–9 (see Section 4.2.6). The quantities of various waste forms generated as a result of Alternative 5 at ANL-W are provided in **Table 4–54**.

Estimates of the total amount of other associated process waste that would be generated are based on an evaluation of waste forecasts from ANL-W that accounted only for the fraction of total ANL-W waste that would be attributable to the treatment of sodium-bonded spent nuclear fuel. The values in Table 4–54 are for disposal and account for volume reduction. It is anticipated that a large fraction of the low-level radioactive waste that would be generated as a result of Alternative 5 could be volume-reduced at the Waste Experimental Reduction Facility at INEEL prior to disposal at the INEEL Subsurface Disposal Area at the Radioactive Waste Management Complex.

The waste values in Table 4–54 are total quantities that would be produced as a result of Alternative 5 operations. They are not incremental increases over the volumes provided in Table 4–10 that would result from the No Action Alternative. In Alternative 5, the driver spent nuclear fuel would be transformed into high-level radioactive waste forms (ceramic and metallic) at ANL-W for disposal in the repository, and in this conversion process, the total volume of material to be disposed of in the repository would be reduced from direct disposal values. The blanket spent nuclear fuel would be cleaned and declad and sent to SRS for melt and dilute processing. The high-level radioactive waste (melt and dilute product) that would be generated from melt and dilute processing at SRS is presented in Table 4–56.

Table 4–54 Amounts of Waste Generated at ANL-W Under Alternative 5[a]

Waste Stream	Waste Quantities	
	Volume (cubic meters)	Mass (kilograms)
Direct Process Waste		
Fuel assembly hardware (low-level radioactive waste)	37.5	13,100
High-level radioactive ceramic waste	16.3 (26 canisters)[b]	24,400
High-level radioactive metallic waste	1.3 (2 canisters)[b]	2,500
Spent nuclear fuel	0	0
Other Associated Process Waste		
High-level radioactive waste	0.4 (1 canister)[b]	220
Low-level radioactive waste[c]	555	113,000
Transuranic waste	9.1	3,800
Mixed waste	27.5	14,800
Sanitary waste	4,960	1.7×10^6
Deactivation Waste		
Low-level radioactive waste[c]	178	60,000
Transuranic waste	1.6	853
Mixed waste	5.1	3,400

[a] These waste generation estimates are through the year 2015. This is the assumed date that these materials might be sent to the repository. Treatment, high-level radioactive waste processing, deactivation, and interim storage would be accomplished during this time period.

[b] Standardized canisters.

[c] The volumes listed represent final disposal volumes following volume reduction at the Waste Experimental Reduction Facility at INEEL.

Source: ANL 1999.

Direct Process Waste

For this alternative, fuel hardware would be removed from the fuel elements in the Fuel Conditioning Facility air cell and disposed of as low-level radioactive waste. These hardware components would be primarily stainless steel materials that contain short-lived radionuclides. This waste stream has been produced at ANL-W for many years and would be handled, as in the past, according to DOE Orders and ANL-W waste management procedures. In addition, the blanket spent nuclear fuel cladding would be included in the fuel hardware stream.

Under this alternative, metallic and ceramic high-level radioactive waste would be a primary product of the electrometallurgical treatment of driver spent nuclear fuel. The salt removed from the electrorefiners would contain the majority of fission products and transuranics from the spent nuclear fuel. This removed salt would be packaged and transferred to the Hot Fuel Examination Facility for processing into ceramic waste. The metallic waste form would consist primarily of stainless steel cladding hulls containing the noble metal fission products. The hulls would be removed from the electrorefiner and packaged for shipment to the Hot Fuel Examination Facility for processing into the metallic waste form. Both of these waste types would be categorized as high-level radioactive waste. The volumes of waste forms provided in Table 4–54 are for the standardized canisters required for disposal of these materials.

The metallic and ceramic high-level radioactive waste generated as a result of electrometallurgical treatment of driver spent nuclear fuel under this alternative, and the driver and blanket spent nuclear fuel from the demonstration project at ANL-W, would be stored temporarily at the Radioactive Scrap and Waste Facility at ANL-W to allow retrieval for future disposal. The Radioactive Scrap and Waste Facility was designed and constructed for temporary storage of this type of waste. Shielding would be provided by a combination

of (1) steel storage liners which would store the waste, and (2) the soil surrounding the liners. When a geologic repository is available, the waste cans containing the metallic and ceramic high-level radioactive waste would be removed from storage, shipped to the INEEL Dry Transfer Facility, and prepared for shipment to the repository.

Other Associated Process High-Level Radioactive Waste

High-level radioactive waste could be generated as a result of blanket spent nuclear fuel processing at ANL-W and SRS. This would result from activities in the Hot Fuel Examination Facility (at ANL-W) and Building 105-L (at SRS). Material in this waste stream would consist of the absorbent used in the off-gas system which has collected the volatile radionuclides released from the spent nuclear fuel when heated.

The volume of high-level radioactive waste that would be generated is expected to be less than the amount needed to fill a single standardized waste canister. Conservatively, the volume of a single canister, 0.4 cubic meters, has been used for the volume of high-level radioactive waste generated.

Other Associated Process Low-Level Radioactive Waste

Low-level radioactive waste would be generated as a result of decladding and cleaning blanket spent nuclear fuel and electrometallurgically treating driver spent nuclear fuel at ANL-W. This would result from activities in the Fuel Conditioning Facility and the Hot Fuel Examination Facility at ANL-W (e.g., equipment decontamination and repair), as well as in other facilities at ANL-W (e.g., analytical laboratory activities). Material in this waste stream has been generated and routinely handled at ANL-W for many years.

The volume of low-level radioactive waste resulting from decladding and cleaning of blanket spent nuclear fuel and electrometallurgical treating driver spent nuclear fuel at ANL-W that would require disposal (after volume reduction) would be approximately 50 cubic meters per year (1,766 cubic feet per year). This represents approximately 1 percent of the total annual volume of low-level radioactive waste currently being disposed of at the INEEL Subsurface Disposal Area at the Radioactive Waste Management Complex, and the total of 555 cubic meters (19,600 cubic feet) represents approximately 0.7 percent of the total Radioactive Waste Management Complex disposal capacity.

Other Associated Process Transuranic Waste

Transuranic waste would be generated from decontamination activities for repair and maintenance of items, and miscellaneous work associated with processing the sodium-bonded spent nuclear fuel. Transuranic waste would be generated primarily from activities conducted in gloveboxes and hot cells at ANL-W.

All of the transuranic waste generated at ANL-W would be packaged and certified in accordance with the Waste Isolation Pilot Plant acceptance criteria prior to transport to the Waste Isolation Pilot Plant facility. The transuranic waste generated would amount to approximately 1 cubic meter per year (35 cubic feet per year), which is less than 0.002 percent of the volume of transuranic waste in retrievable storage at the Radioactive Waste Management Complex at INEEL. The total volume of transuranic waste would be approximately 9.1 cubic meters (321 cubic feet), which is approximately 0.005 percent of the estimated total volume of transuranic waste to be placed at the Waste Isolation Pilot Plant.

Other Associated Process Mixed Waste

Mixed waste would be generated primarily from the disposal of any cadmium-contaminated equipment or cleanup material and the analysis of cadmium samples. Mixed waste would be handled according to ANL-W procedures that require limited accumulation at the point of generation. Interim storage of this waste would

be accomplished at the Radioactive Scrap and Waste Facility prior to eventual disposal. The Radioactive Scrap and Waste Facility is a permitted mixed waste storage facility for these materials. The mixed waste streams that contribute to the overall mixed waste generated by electrometallurgical treatment have been identified in the INEEL Site Treatment Plan (DOE 1995b).

Deactivation Waste

A variety of waste would be generated as part of deactivation activities at ANL-W. This would include process equipment and process material such as electrorefiner salt and cadmium from electrometallurgical treatment of driver spent nuclear fuel. Generated waste categories would include low-level radioactive waste, transuranic waste, and mixed waste. This waste would be categorized and disposed of according to DOE Orders and ANL-W radioactive waste management procedures, as described above for each waste category.

The largest volume of deactivation waste would be low-level radioactive waste generated as a result of equipment dismantling and disposal. Components of electrometallurgical treatment of the driver spent nuclear fuel that would require disposition include two electrorefiners, two hot hydrostatic presses, and two V-mixers, as well as other components such as the grinder/crusher. Deactivation of these components would generate additional mixed, transuranic, and low-level radioactive waste that would require management.

The deactivation waste volume would be generated over a period of two years. The total deactivation waste would represent an additional 30 percent over the total associated process waste requiring disposal.

SRS

The melt and dilute process at SRS would generate process waste from treatment operations and other associated process waste from support operations. Process waste would include metallic high-level radioactive waste. Other associated process waste would include operational waste such as failed equipment, rags, packaging materials, and other miscellaneous items. The associated process waste would include low-level radioactive waste, transuranic waste, and mixed waste. All of the waste streams would be categorized according to existing DOE Orders and SRS waste management procedures. The anticipated categorization of the waste types and their expected interim storage and final disposal locations are given in **Table 4–55**.

Table 4–55 Waste Material Categories at SRS and Interim and Final Locations

Waste Stream	*Category*	*Interim Storage Location*	*Final Disposal Location*
Process Waste			
Melt and dilute product	High-level radioactive metallic waste	L-Area	Geologic repository
Off-gas filters	High-level radioactive waste [a]	L-Area	Geologic repository
Other Associated Process Waste			
Less than 100 nanocuries per gram transuranic waste Contaminated	Low-level radioactive waste	None	Low-activity waste vaults
Greater than 100 nanocuries per gram transuranic waste	Transuranic waste	Transuranic waste storage pads	Waste Isolation Pilot Plant
Offsite	Mixed waste	Mixed waste storage buildings	Offsite

[a] Cleaning of the contaminated filters would generate high-level radioactive liquid waste.

Estimates of the amounts of waste generated as a result of the melt and dilute processing at SRS are provided in **Table 4–56**. These values are based on an evaluation of waste forecasts that accounts only for the fraction of total waste that would be attributable to processing the blanket spent nuclear fuel pins.

Table 4–56 Amounts of Waste Generated at SRS Under Alternative 5

Waste Stream	*Waste Quantities (cubic meters)* [a]
Direct Process Waste	
Melt and dilute product	76 (189 canisters) [b]
Liquid high-level radioactive waste	30 [c]
Saltstone [d]	78
Other Associated Process Waste	
Low-level radioactive waste	330 [e]
Transuranic waste	16.5
Mixed waste	3

[a] Except for the number of canisters of melt and dilute products, the values given are estimated based on the heavy metal mass ratio of similar material processed at SRS (20 metric tons of heavy metal) and provided in DOE 2000.
[b] Standardized spent nuclear fuel canisters.
[c] This is a liquid high-level radioactive waste volume which results in about one Defense Waste Processing Facility borosilicate glass high-level radioactive waste canister or a solid high-level radioactive waste volume of 0.62 cubic meters.
[d] This is a secondary process waste from processing the high-level radioactive waste.
[e] Final disposal volume following a volume reduction (a reduction factor of 4 was assumed).

As indicated in the following waste-type discussions, the amounts of waste associated with this processing alternative are relatively small compared to onsite and offsite management capacities.

Direct Process Waste

During the melt and dilute process, a high-level radioactive waste melt and dilute product (metallic waste) would be the primary product. This waste would be temporarily stored in L-Area prior to ultimate disposition in an offsite (proposed geologic) repository. In addition, some high-level radioactive waste would be generated from cleaning the off-gas filter system, which contains cesium, tellurium, and other isotopes volatilized during the melt and dilute process. The high-level radioactive waste would be processed in the Defense Waste Processing Facility to yield vitrified high-level radioactive waste and saltstone. The vitrified high-level radioactive waste would be temporarily stored at the Defense Waste Processing Facility pending ultimate disposal in a geologic repository. The saltstone is a cement form of low-level radioactive waste that is generated as a by-product of SRS high-level radioactive waste tank form operations. The saltstone would be disposed of on site in the Z-Area Saltstone Vaults. The volume of this saltstone would be about 0.0070 percent of the 1.11 million-cubic meter (39.2 million-cubic foot) disposal capacity of the low-activity waste vaults.

Other Associated Process Low-Level Radioactive Waste

Low-level radioactive waste would be generated during the melt and dilute process. The volume of low-level radioactive waste resulting from this alternative (after volume reduction) would be about 1.1 percent of the total 30,500-cubic meter (1.08 million-cubic foot) disposal capacity of the low-activity waste vaults.

Other Associated Process Transuranic Waste

The volume of transuranic waste that would be generated during the melt and dilute process would be about 0.01 percent of the current 168,500-cubic meter (5.95 million-cubic foot) limit for the Waste Isolation Pilot Plant (DOE 1997).

Other Associated Process Mixed Waste

Mixed waste that would be generated during the melt and dilute process would be temporarily stored on site in the Mixed Waste Storage Buildings prior to eventual offsite disposal. The volume of this waste would be about 0.16 percent of the 1,900-cubic meter (67,100-cubic foot) storage capacity of these storage buildings.

4.8 ALTERNATIVE 6: MELT AND DILUTE BLANKET AND DRIVER FUEL AT ANL-W

Under this alternative, sodium-bonded blanket and driver spent nuclear fuel would be treated using the melt and dilute process at ANL-W. The melt and dilute product generated by this treatment process would be stored at the Radioactive Scrap and Waste Facility pending repackaging and transportation for disposal in a geologic repository. Both blanket and driver spent nuclear fuel would be cleaned to remove metallic sodium to the extent possible. The removed sodium would be stabilized using an oxidation/carbonation process (ANL 1999). The preparation of driver and blanket spent nuclear fuel to remove metallic sodium could start in 2003. The treatment of driver and blanket spent nuclear fuel by melt and dilute processing at ANL-W could start in 2005 and could be completed by 2015. In addition, a full year of operations would be needed to deactivate the processing equipment and establish an industrially safe configuration.

4.8.1 Air Quality

Nonradiological Gaseous Emissions

It is expected that criteria and hazardous air pollutants released from operational activities at ANL-W under this alternative would be the same as for Alternative 1, with the exception that there would be no cadmium release, as described in Section 4.3.1 (see also Appendix E, Section E.5.3.1, for more detail). Baseline air quality concentrations are presented in Section 3.2.3.1.

Radiological Gaseous Emissions

The cleaning of the blanket and driver spent nuclear fuel and the melt and dilute treatment of this fuel would release gaseous fission products to the hot argon cell environment. Krypton-85 and elemental tritium would be the most prevalent radioactive gaseous fission products released to the environment. The tritium released into the cell would not be oxidized due to a very low presence of oxygen and humidity in the argon cell. The argon cell also would contain an equilibrium concentration of other radionuclide isotopes. Appendix E, Section E.4.1, provides a list of various isotopes that would be present in the argon cell in nanocuries (10^{-9} curies) and would be released to the atmosphere through the facility stack, along with krypton-85 and elemental tritium. The maximum release of radioactive gases would occur when cutting of blanket and driver spent nuclear fuel to remove metallic sodium is performed simultaneously. This simultaneous operation could occur over a 2-year period during the estimated 10 years of operation, starting in 2003. Appendix E, Section E.4.5, provides more details on various releases during the processing period at ANL-W. Based on an annual processing throughput of 10 metric tons of heavy metal of blanket spent nuclear fuel elements and about 1.7 metric tons of heavy metal of driver spent nuclear fuel elements, about 2,162 curies of elemental tritium and 32,650 curies of gaseous krypton-85 would be released annually to the atmosphere (see Appendix E, Section E.4.5).

4.8.2 Water Resources

Surface Water

No surface water is used at ANL-W. Flood waters from the Big Lost River would not be expected to reach the facilities at ANL-W as shown in Figure 3–3.

Nonradiological Liquid Effluent

There are no discharges to the surface waters at ANL-W, except for discharges of nonhazardous liquid waste to the sewage pond and the industrial waste pond. Big Lost River, Little Lost River, and Birch Creek would not be impacted by activities associated with the melt and dilute treatment process. Current operating and monitoring practices would continue for stormwater and liquid effluent discharges associated with facilities at ANL-W (see also Section 4.2.2).

During fuel treatment and associated activities, some hazardous materials may be used inside buildings. To prevent potential releases to surface or subsurface waters resulting from spills of hazardous materials used in buildings, these facilities are designed, constructed, and maintained to contain these materials. Double-contained pipes, leak detection, and secondary containment of tanks are some of the features used to prevent hazardous materials from being released to the environment. Following existing written procedures, spill containment and cleanup equipment is present in areas where hazardous materials are stored or used (DOE 1996b).

Radiological Liquid Effluent

No radiological liquid effluent generated by melt and dilute treatment process operations would be discharged to surface water at ANL-W.

Groundwater

Under this alternative at ANL-W, there would be little change in groundwater consumption for domestic use since there is little change expected in the number of workers. Water consumption for the melt and dilute treatment process operations would not impact the current water usage at ANL-W. The current water usage at ANL-W is 188 million liters per year (49.6 million gallons per year).

Nonradiological Liquid Effluent

No nonradiological liquid effluent generated by melt and dilute treatment process operations would be discharged to groundwater.

Radiological Liquid Effluent

No radiological liquid effluent generated by melt and dilute treatment process operations would be discharged to groundwater.

4.8.3 Socioeconomics

Under this alternative, the existing facilities at ANL-W would remain operational. No new employment or in-migration of workers would be required. Thus, there would be no additional impacts on the socioeconomic conditions in the region around INEEL.

4.8.4 Public and Occupational Health and Safety

The assessments of potential incremental radiological and chemical impacts associated with this alternative are presented in this section. Summaries of radiological impacts from normal operations are presented in Tables 4–57 and 4–58 for the public and workers, respectively. The radiological impacts from a spectrum of hypothetical accident scenarios are provided in Tables 4–59 and 4–60. The impacts from hazardous

chemical releases during accident conditions are presented in Table 4–61. Background information on the effects of radiation on human health and safety is presented in Appendix E, Section E.2.

4.8.4.1 Normal Operations

Radiological Impacts

Under this alternative, radioactive releases would occur during sodium-bonded blanket spent nuclear fuel and driver spent nuclear fuel cleaning and melt and dilute processes. All of these activities would be performed in the argon cell. Appendix E, Sections E.3, E.4.1, and E.4.2, details the method and assumptions used for calculating the impacts of normal operational radiological releases on the public health and safety. The maximum annual dose to the public would result when both blanket and driver spent nuclear fuel are treated simultaneously under this alternative. Appendix E, Section E.4.5, provides the details on treatment process duration and throughputs for each fuel type.

Calculated maximum annual and project total radiological impacts to the public are given in **Table 4–57**. The impacts are calculated for two types of receptors: the general public living within 80 kilometers (50 miles) of ANL-W in the year 2010, and a maximally exposed offsite individual (a member of the public assumed to be residing at the INEEL site boundary and receiving the maximum dose). Primary contributors to doses to members of the public would be from releases of tritium gases (about 1 percent of which were assumed conservatively to be in oxidized form) and krypton-85; together they contribute over 99.9 percent of the total calculated doses. To put the operational impacts into perspective, comparisons with impacts from natural background radiation also are included in the table. As shown in this table, the expected radiation doses to the maximally exposed offsite individual and the general public are much smaller than the limit of 10 millirem per year set by the EPA (40 CFR 61) and DOE (DOE Order 5400.5).

Table 4–58 summarizes worker population doses. Occupational doses were estimated by examining the type and duration of various operations performed by the ANL-W workers involved with the melt and dilute treatment processes. It was concluded that the average worker dose would not be different from that currently being experienced. The estimated annual collective worker dose would be 22 person-rem, with an average individual dose of 60 millirem per year for each of the 346 involved workers. If these estimates were extended over the 12 years of treatment activities (assuming operations would start in 2003 and end in 2015) and the 33 person-rem from 1 year of deactivation activities were included, the project total worker population dose would be 297 person-rem, leading to a risk of 0.12 latent cancer fatalities.

As shown in Tables 4–57 and 4–58:

- The annual dose to the maximally exposed offsite individual would be 0.0020 millirem per year, with an associated risk of developing a fatal cancer of 1×10^{-9} per year (or one chance in 1 billion that the individual would develop a fatal cancer per year of exposure).

- The collective dose to the population within 80 kilometers (50 miles) of the ANL-W facilities would be 0.012 person-rem per year, with an associated 6×10^{-6} latent cancer fatalities per year (or one chance in 167,000 that the population would develop a fatal cancer per year of exposure).

- The project total risk to the population within 80 kilometers (50 miles) of the ANL-W facilities would be 0.000012 latent cancer fatalities (or one chance in 83,000 that the exposed population would develop a fatal cancer).

Table 4–57 Annual and Project Total Radiological Impacts to the Public From Operational Activities Under Alternative 6

Receptor	*Melt and Dilute Driver Spent Nuclear Fuel at ANL-W*	*Melt and Dilute Blanket Spent Nuclear Fuel at ANL-W*	*Total*
Population Within 80 Kilometers (50 Miles) in the Year 2010			
Collective dose (person-rem) [a]	0.012	0.00028	0.012
Excess latent cancer fatalities (per year)	6.0×10^{-6}	1.4×10^{-7}	6.1×10^{-6}
Project total excess latent cancer fatalities [b]	0.000012	3.4×10^{-7}	0.000012
Maximally Exposed Offsite Individual			
Dose (millirem per year) [a]	0.002	0.000048	0.0020
Percent of annual background radiation [c]	0.00056	0.000013	0.00057
Latent cancer fatality risk (per year)	1.0×10^{-9}	2.4×10^{-11}	1.0×10^{-9}
Project total lifetime cancer fatality risk	2.0×10^{-9}	5.8×10^{-11}	2.0×10^{-9}
Average Individual Within 80 Kilometers (50 Miles)			
Dose (millirem per year) [d]	0.00005	1.2×10^{-6}	0.000051
Latent cancer fatality risk (per year)	2.5×10^{-11}	5.8×10^{-13}	2.6×10^{-11}
Project total lifetime cancer fatality risk [b]	5.0×10^{-11}	1.4×10^{-12}	5.1×10^{-11}

[a] Annual maximum dose during normal operations.
[b] Total calculated dose over 12 years. Nearly all of the impacts are associated with releases of tritium, krypton, and iodine that would occur during the cleaning process. The impact of releases resulting from melt and dilute processing only are not significant.
[c] The annual natural background radiation level at INEEL is about 360 millirem for the average individual (see Table 3–8); the population within 80 kilometers (50 miles) in the year 2010 would receive 86,500 person-rem.
[d] Obtained by dividing the population dose by the number of people projected to live within 80 kilometers (50 miles) of ANL-W in the year 2010 (240,338).

Table 4–58 Annual and Project Total Radiological Impacts to Workers From Operational Activities Under Alternative 6

Receptor	*Impacts*
Worker [a]	
Average worker dose (millirem per year)	60
Average worker latent cancer fatality risk (project total over 12 years)	0.00029
Worker Population	
Collective dose (person-rem per year) [b]	22
Excess latent cancer fatalities (per year)	0.0088
Project total dose (person-rem)	297
Project total excess latent cancer fatalities	0.12

[a] The regulatory dose limit for an individual worker is 5,000 millirem per year (10 CFR 835). However, the maximum annual dose to a worker would be kept below the DOE Administrative Control Level of 2,000 millirem per year, as established for all DOE activities in DOE Order N 441.1.
[b] Increases to 33 person-rem for one year of deactivation activities.

Source: ANL 1999.

- The collective dose to facility workers would be 22 person-rem per year, with an associated 0.0088 latent cancer fatalities (or one chance in 113 that the workers would experience a fatal cancer per year of operation).

- The project total dose to facility workers would be 297 person-rem with an associated 0.12 latent cancer fatalities (or one chance in eight that the exposed workers would experience a fatal cancer).

These results indicate that no increase in the expected number of fatal cancers among the various affected populations would result from the activities under this alternative.

Hazardous Chemical Impacts

The hazardous chemical impacts to the public residing at the INEEL site boundary from the operational activities at ANL-W under Alternative 6 would be similar to the impacts evaluated for Alternative 1, with the exception that there would be no cadmium release, as described in Section 4.3.4.1. The results indicate that no adverse toxic health or cancer effects would be expected from exposure to hazardous chemicals released under this alternative. The existing chemical environment is presented in Section 3.2.10.2.

4.8.4.2 Facility Accidents

Radiological Impacts

The potential radiological impacts to the public and a noninvolved onsite worker due to accidents during operational activities related to melt and dilute processing of fuel elements are summarized and presented in this section. The detailed analysis of facility accidents and the associated assumptions are presented in Appendix F. The detailed analysis considered a wide spectrum of potential accident scenarios, including fire, spills, criticality, earthquake, and aircraft crash. The aircraft crash event was determined to have an occurrence frequency of less than 10^{-7} per year, and consequence analyses for this event were not performed. Double-batching of the driver spent nuclear fuel was determined potentially to result in a criticality event (see Appendix F), and this event was analyzed for the driver spent nuclear fuel only. Processing of the blanket and driver spent nuclear fuel would be performed in the Hot Fuel Examination Facility. The multifacility impacts of the beyond-design-basis earthquake are not relevant to processing of the fuel under this option. Therefore, only the higher frequency design-basis earthquake event was analyzed. **Table 4–59** presents the frequencies and consequences of the postulated set of accidents to the maximally exposed offsite individual; the offsite population residing within 80 kilometers (50 miles) of the facility; and a noninvolved worker located 100 meters (330 feet) to 230 meters (755 feet) from the facility. The 230-meter (755-foot) distance is the ANL-W bus staging area, which leads to a higher dose to the noninvolved worker for the scenarios with elevated releases.

The dose to the maximally exposed offsite individual was calculated for the 95th percentile meteorological conditions. The doses to the population and the noninvolved worker were calculated using 50th percentile meteorological conditions. DOE did not quantitatively estimate the involved worker dose due to accidents. (See the discussion on the involved worker in Section 4.2.4.2.) The accident risks for the same receptors are summarized in **Table 4–60**.

For accidents at ANL-W, the highest risk of a latent cancer fatality to the maximally exposed offsite individual and a noninvolved worker would be 0.000076 per year (or one chance in 13,160 that the individual would develop a fatal cancer per year of operation) and 2.7×10^{-6} per year (or one chance in 370,400 that the individual would develop a fatal cancer per year of operation), respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.0090 per year (or one chance in 111 that the population would experience a fatal cancer per year of operation).

Table 4–59 Accident Frequency and Consequences Under Alternative 6

Accident	Frequency (event per year)	Maximally Exposed Offsite Individual		Population Within 80 Kilometers (50 Miles)		Noninvolved Worker	
		Dose (millirem)	Latent Cancer Fatality Risk [a]	Dose (person-rem)	Excess Latent Cancer Fatalities [b]	Dose (millirem)	Latent Cancer Fatality Risk [a]
Driver Spent Nuclear Fuel							
Waste handling spill	0.0024	597	0.00030	70.8	0.035	26.7	0.000011
Transuranic waste fire	0.001	0.059	3.0×10^{-8}	0.0071	3.6×10^{-6}	0.22	8.8×10^{-8}
Cask drop	0.01	0.030	1.5×10^{-8}	0.0035	1.8×10^{-6}	0.00084	3.4×10^{-10}
Design-basis earthquake	0.008	19000	0.0095	2250	1.1	840	0.00034
Sodium fire[c]	0.008	282	0.00014	33	0.016	2.6	1.0×10^{-6}
Criticality	0.0003	0.52	2.6×10^{-7}	0.085	0.000043	0.47	1.9×10^{-7}
Blanket Spent Nuclear Fuel							
Waste handling spill	0.0024	15	7.5×10^{-6}	1.8	0.00090	0.49	2.0×10^{-7}
Transuranic waste fire	0.001	0.059	3.0×10^{-8}	0.0071	3.6×10^{-6}	0.22	8.8×10^{-8}
Cask drop	0.01	0.0024	1.2×10^{-9}	0.00028	1.4×10^{-7}	0.000049	2.0×10^{-11}
Design-basis earthquake	0.008	471	0.00024	56.1	0.028	15.2	6.1×10^{-6}
Sodium fire [c]	0.008	5.9	3.0×10^{-6}	0.69	0.00035	0.054	2.2×10^{-8}

[a] Increased likelihood of a latent cancer fatality.
[b] Increased number of latent cancer fatalities.
[c] The frequency for this event is the frequency for the facility design-basis earthquake-initiating cell fire.

Table 4–60 Annual Cancer Risks Due to Accidents at ANL-W Under Alternative 6

Accident	Maximally Exposed Offsite Individual [a]	Population Within 80 Kilometers (50 Miles) [b]	Noninvolved Worker [a]
Driver Spent Nuclear Fuel			
Waste liquid spill	7.2×10^{-7}	0.000085	2.6×10^{-8}
Transuranic waste fire	3.0×10^{-11}	3.6×10^{-9}	8.8×10^{-11}
Cask drop	1.5×10^{-10}	1.7×10^{-8}	3.4×10^{-12}
Design-basis earthquake	0.000076	0.0090	2.7×10^{-6}
Sodium fire	1.1×10^{-6}	0.00013	8.3×10^{-9}
Criticality	8.0×10^{-11}	1.3×10^{-8}	5.7×10^{-11}
Blanket Spent Nuclear Fuel			
Waste liquid spill	1.8×10^{-8}	2.2×10^{-6}	4.8×10^{-10}
Cask drop	1.2×10^{-11}	1.4×10^{-9}	2.0×10^{-13}
Transuranic waste fire	3.0×10^{-11}	3.6×10^{-9}	8.8×10^{-11}
Design-basis earthquake	1.9×10^{-6}	0.00023	4.8×10^{-8}
Sodium fire	2.4×10^{-8}	2.8×10^{-6}	1.7×10^{-10}

[a] Increased likelihood of a latent cancer fatality.
[b] Increased number of latent cancer fatalities.

Hazardous Chemical Impacts

Nonradiological impacts are evaluated in terms of comparison to ERPG. ERPG values are estimates of airborne concentration thresholds above which one can reasonably anticipate observing adverse effects (see Appendixes F, Section F.3.1.2 for details).

The hazardous chemical impacts of potential facility accidents associated with the treatment of the driver spent nuclear fuel using the electrometallurgical process are summarized in **Table 4–61**.

Table 4–61 Hazardous Chemical Impacts Due to Accidents at ANL-W Under Alternative 6

Accident	*Frequency (event per year)*	*Receptor*	*Exposure*
Sodium fire	0.008	Noninvolved worker	Sodium: less than ERPG-1
		Maximally exposed offsite individual	Sodium: less than ERPG-1

ERPG = Emergency Response Planning Guideline.

4.8.5 Environmental Justice

As discussed in Appendix H, Executive Order 12898 directs Federal agencies to address disproportionately high and adverse health or environmental effects of alternatives on minority or low-income populations.

Analyses of normal operations and accident conditions presented in Section 4.8.4 show the risk of latent cancer fatalities to the public residing within 80 kilometers (50 miles) of the melt and dilute treatment processing facilities at ANL-W to be much lower than 1. Therefore, radiological and nonradiological risks posed by implementation of this alternative would have no disproportionately high and adverse consequences on any particular group within the general population, including minority or low-income populations.

4.8.6 Waste Management

This alternative would generate process waste from treatment operations, other associated process waste from normal support operations, and deactivation waste following the conclusion of operations. Process waste would include high-level radioactive metallic and ceramic waste from stabilizing the residual waste from the existing Electrometallurgical Demonstration Project. Other associated process waste would include operational waste such as failed equipment, rags, packaging materials, and other miscellaneous items. Deactivation waste would include the disposal of process equipment and other materials. All of these materials would be categorized according to existing DOE Orders and ANL-W waste management procedures. The anticipated categorization of these waste types and their expected interim storage and final disposal locations are given in Table 4–9 (see Section 4.2.6). The quantities of various waste forms generated as a result of Alternative 6 are provided in **Table 4–62**.

Estimates of the total amount of other associated process waste generated are based on an evaluation of waste forecasts from ANL-W, together with an understanding of melt and dilute process activities resulting in the generation of each waste category. The values in Table 4–62 are for disposal and include volume reduction. It is anticipated that a large fraction of the low-level radioactive waste that would be generated as a result of Alternative 6 could be volume-reduced by up to 100 percent at the Waste Experimental Reduction Facility at INEEL prior to disposal at the INEEL Subsurface Disposal Area at the Radioactive Waste Management Complex.

Table 4–62 Amounts of Waste Generated at ANL-W Under Alternative 6[a]

Waste Stream	*Waste Quantities*	
	Volume (cubic meters)	*Mass (kilograms)*
Direct Process Waste		
Fuel assembly hardware (low-level radioactive waste)	0	0
High-level radioactive ceramic waste	19.4 (31 canisters) [b]	29,000
High-level radioactive metallic waste	0.6 (1 canister) [b]	460
Melt and dilute product	65.6 (164 canisters) [b]	136,400
Other Associated Process Waste		
High-level radioactive waste	0.4 (1 canister) [b]	220
Low-level radioactive waste [c]	711	144,000
Transuranic waste	12.5	5,400
Mixed waste	35.3	19,000
Sanitary waste	4,960	1.72×10^{6}
Deactivation Waste		
Low-level radioactive waste [c]	213	72,000
Transuranic waste	1.6	853
Mixed waste	5.9	3,500

[a] These waste generation estimates are through the year 2015. This is the assumed date that these materials might be sent to the repository. Treatment, high-level radioactive waste processing, deactivation, and interim storage would be accomplished during this time period.

[b] Standardized canisters.

[c] The volumes listed represent final disposal volumes following volume reduction at the Waste Experimental Reduction Facility at INEEL.

Source: ANL 1999.

The waste values in Table 4–62 represent total quantities that would be produced as a result of Alternative 6 operations. They are not incremental increases over the volumes provided in Table 4–10 that would result from the No Action Alternative. In Alternative 6, both the driver and blanket spent nuclear fuel would be transformed into a high-level radioactive waste form (melt and dilute product) for disposal in the repository, and in this conversion process, the total volume of material to be disposed of in the repository would be reduced from direct disposal values.

Direct Process Waste

For this alternative, fuel hardware would be used as part of the stainless steel to form the metal ingot for disposal of the fuel by melting. Its mass is included as part of the spent nuclear fuel disposal.

Under this alternative, metallic and ceramic high-level radioactive waste would be produced from existing process material at ANL-W. This waste would be generated to stabilize materials produced during the electrometallurgical demonstration project. In addition, the salt removed from the melting furnace used for driver spent nuclear fuel would contain fission products that would be stabilized in ceramic waste. The volumes of waste forms provided in Table 4–62 are for the standardized canisters required for disposal of these materials.

A second metallic high-level radioactive waste called the melt and dilute product would be generated as a result of the melt and dilute treatment of fuel at ANL-W. This waste, along with the ceramic and metallic waste from the demonstration project, would be stored temporarily at the Radioactive Scrap and Waste Facility at ANL-W to allow retrieval for future disposal. The Radioactive Scrap and Waste Facility was designed and constructed for temporary storage of this type of waste. Shielding would be provided by a

combination of (1) steel storage liners which would store the waste, and (2) the soil surrounding the liners. When a geologic repository is available, the waste cans containing this high-level radioactive waste would be removed from storage, shipped to the INEEL Dry Transfer Facility, and prepared for shipment to the repository.

Other Associated Process High-Level Radioactive Waste

High-level radioactive waste would be generated as a result of driver and blanket spent nuclear fuel processing at ANL-W. This would result from activities in the Hot Fuel Examination Facility. Material in this waste stream would consist of the absorbent used in the off-gas system which would have collected the volatile radionuclides released from the spent nuclear fuel when heated.

The volume of high-level radioactive waste generated is expected to be less than the amount needed to fill a single standardized waste canister. Conservatively, the volume of a single canister, 0.4 cubic meters, was used for the volume of high-level radioactive waste generated.

Other Associated Process Low-Level Radioactive Waste

Low-level radioactive waste would be generated as a result of the melt and dilute treatment of fuel at ANL-W. This would result from activities in the Fuel Conditioning Facility and the Hot Fuel Examination Facility at ANL-W (e.g., equipment decontamination and repair), as well as in other facilities at ANL-W (e.g., analytical laboratory activities). Material in this waste stream has been generated and routinely handled at ANL-W for many years.

The volume of low-level radioactive waste at ANL-W that would require disposal (after volume reduction) would be approximately 50 cubic meters (1,766 cubic feet) per year. This represents approximately 1 percent of the total annual volume of low-level radioactive waste currently being disposed of at the INEEL Subsurface Disposal Area at the Radioactive Waste Management Complex, and the total of 711 cubic meters (25,100 cubic feet) represents approximately 0.6 percent of the total Radioactive Waste Management Complex disposal capacity.

Other Associated Process Transuranic Waste

Transuranic waste would be generated from decontamination activities for repair and maintenance of items, and miscellaneous work associated with processing the sodium-bonded spent nuclear fuel. Transuranic waste would be generated primarily from activities conducted in gloveboxes and hot cells at ANL-W.

All of the transuranic waste generated at ANL-W would be packaged and certified in accordance with Waste Isolation Pilot Plant acceptance criteria prior to transport to the Waste Isolation Pilot Plant. The transuranic waste generated would amount to approximately 1 cubic meter (35 cubic feet) per year, which is less than 0.002 percent of the volume of transuranic waste in retrievable storage at the Radioactive Waste Management Complex at INEEL. The total volume of transuranic waste would be 12.5 cubic meters (441 cubic feet), which is less than 0.008 percent of the estimated total volume of transuranic waste to be placed at the Waste Isolation Pilot Plant.

Other Associated Process Mixed Waste

Mixed waste would be generated primarily from the disposal of any cadmium-contaminated equipment or cleanup material and the analysis of cadmium samples. Mixed waste would be handled according to ANL-W procedures that require limited accumulation at the point of generation. Interim storage of this waste would be accomplished at the Radioactive Scrap and Waste Facility prior to eventual disposal. The Radioactive

Scrap and Waste Facility is a permitted mixed waste storage facility for these materials. The mixed waste streams that contribute to the overall mixed waste generated by electrometallurgical treatment are identified in the INEEL Site Treatment Plan (DOE 1995b).

Deactivation Waste

A variety of waste would be generated as part of deactivation activities associated with melt and dilute treatment of sodium-bonded spent nuclear fuel at ANL-W. Generated waste categories would include low-level radioactive waste, transuranic waste, and mixed waste. This waste would be categorized and disposed of according to DOE Orders and ANL-W radioactive waste management procedures, as described above for each waste category.

The largest volume of deactivation waste would be low-level radioactive waste generated as a result of dismantling and disposing of electrometallurgical treatment and melt and dilute processing equipment. Components of the electrometallurgical demonstration project that would require disposition include two electrorefiners; two hot hydrostatic presses; and one V-mixer, as well as other components such as the grinder/crusher. Deactivation of components would generate additional mixed, transuranic, and low-level radioactive waste that would require management. The total deactivation waste would represent 35 percent over the total associated process waste requiring disposal.

4.9 TRANSPORTATION IMPACTS

Transportation impacts may be divided into two parts: the impacts of incident-free or routine transportation, and the impacts of transportation accidents. Incident-free transportation and transportation accident impacts are divided into two parts: nonradiological impacts and radiological impacts. Incident-free transportation impacts include radiological impacts on the public and the crew from the radiation field that surrounds the package. Nonradiological impacts of incident-free transportation are from vehicular emissions. Nonradiological impacts of potential transportation accidents include traffic accident fatalities. Only in the worst conceivable conditions, which are of low probability, could a transportation cask of the type used to transport radioactive material be so damaged that a release of radioactivity to the environment could occur.

The impact of a specific accident is expressed in terms of probabilistic risk, which is the probability of that accident occurring multiplied by its consequence. Hypothetical accidents ranging from a low-speed impact to those involving high-speed impacts with or without fires leading to cask failure are analyzed. The accident frequencies and consequences are evaluated using the method developed for the NRC, which is known as the "Modal Study" (NRC 1987). The overall risk is obtained by summing the individual risks from all accident conditions. The risks for radiological accidents are expressed as additional latent cancer fatalities and as additional immediate fatalities for nonradiological accidents. The risks of incident-free effects are expressed in additional latent cancer fatalities.

The first step in the ground transportation analysis was to determine the incident-free and accident risk factors on a per-shipment basis for transportation of the various materials. Calculation of risk factors was accomplished by using the HIGHWAY (Johnson et al. 1993) computer codes to choose representative routes according to U.S. Department of Transportation regulations. These codes provide population estimates so that RADTRAN 5 (Neuhauser and Kanipe 1998) codes could be used to determine the radiological risk factors. This analysis is discussed in Appendix G. **Table 4–63** lists the sodium-bonded spent nuclear fuel destinations evaluated in this EIS.

Table 4–63 Transportation Summary for Sodium-Bonded Fuel

Fuel Type	*Alternatives* [a]	*Metric Tons of Heavy Metal*	*Origin/State*	*Destination/ State*	*Cask*	*Number of Shipments/Type of Transport*
EBR-II driver	All	1.1	ANL-W/ID	ANL-W/ID	HFEF-5	84/On site, intrafacility transfers
EBR-II driver	All	2.0	INTEC/ID	ANL-W/ID	TN-FSV or NAC-LWT	17/On site with roads open or 43/On site with roads open
EBR-II blanket	All	22.4	ANL-W/ID	ANL-W/ID	HFEF-5	165/On site, intrafacility transfers
Fast Flux Test Facility driver [b]	All	0.33	Hanford/WA	ANL-W/ID	T-3	10/Public highways
Fermi-1 blanket	All	34.2	INTEC/ID	ANL-W/ID	PB-1	14/On site with road closed
Miscellaneous [b]	All	0.1	Oak Ridge National Laboratory/TN, Sandia National Laboratories/NM, SRS/SC	ANL-W/ID	To be determined by DOE	1/Public highways 1/Public highways 1/Public highways
Declad EBR-II blanket	3 and 5	22.4	ANL-W/ID	SRS/SC	NAC-LWT	11/Public highways
Declad Fermi-1 blanket	3 and 5	34.2	ANL-W/ID	SRS/SC	NAC-LWT	18/Public highways

ID = Idaho; NM = New Mexico; SC = South Carolina; TN = Tennessee; WA = Washington

[a] "All" includes the proposed action plus the No Action Alternative.

[b] This fuel is assumed to be in Idaho per the amended Record of Decision for the Programmatic Spent Nuclear Fuel EIS (61 FR 9441).

Transportation of the Fast Flux Test Facility driver spent nuclear fuel currently stored at the Hanford site and the small amounts of miscellaneous sodium-bonded spent nuclear fuel currently stored at Oak Ridge National Laboratory, Sandia National Laboratories, and at SRS (see Section 2.2.3 and Appendix D, Section D.5.2, for more details on miscellaneous fuel types) are shipment campaigns related to sodium-bonded spent nuclear fuel and were analyzed by DOE in the Programmatic Spent Nuclear Fuel EIS (DOE 1995a and 61 FR 9441), so they are included by reference in this impact analysis. See Appendix G for more details.

All EBR-II blanket and some EBR-II driver spent nuclear fuel are currently stored at ANL-W and would be subject to a building-to-building movement for processing. Since the movement is a short distance on closed DOE-controlled roads, DOE procedures and the NRC regulations do not require the use of a certified Type B cask. No incident-free risk analysis is necessary because the public would receive no measurable exposure. The worker dose is included in the process and handling dose estimates because the same personnel would be moving the spent nuclear fuel. The probability and consequences of potential accidents during movement are bounded in frequency and consequence by handling accidents.

Fermi-1 blanket spent nuclear fuel would be shipped from INTEC to ANL-W in the Type B cask (PB-1 Cask). Since DOE would close the roads between INTEC and ANL-W using existing traffic gates, and the road is uninhabited, no quantitative analysis is necessary. No incident-free risk analysis is necessary because the public would receive no measurable exposure. The worker dose is included in the process and handling dose estimates because the same personnel would be moving the spent nuclear fuel. Once the cask is closed

for movement on the closed roads, the likelihood and consequences of any foreseeable accident would be very small and are not further quantified.

EBR-II driver spent nuclear fuel would be shipped from INTEC to ANL-W in a certified Type B cask, either model TN-FSV or model NAC-LWT. Since the cask would be certified, DOE would not close the roads between INTEC and ANL-W. However, since the road is uninhabited, limited quantitative analysis is necessary. No incident-free risk analysis for exposure to the public at stops or in their homes is necessary. The worker dose is analyzed for the transportation crew, and the dose to other vehicles using the road is estimated. No accident analysis is necessary because potential accidents during movement are bounded in frequency and consequence by the handling accidents. Once the cask is closed for movement on the closed roads, the likelihood and consequences of any foreseeable accident would be very small.

Transportation-related risks are calculated and presented separately for workers and members of the general public. The workers considered are truck crew members involved in the actual overland transportation. The general public includes all persons who could be exposed to a shipment while it is moving or stopped during transit. The affected population includes individuals living within 800 meters (0.5 miles) of each side of the road. Potential risks are estimated for the affected population and for the hypothetical maximally exposed offsite individual. For incident-free operations, the maximally exposed individual would be an individual stuck in traffic next to the shipment for 30 minutes. For accident conditions, the maximally exposed individual is assumed to be an individual located 33 meters (105 feet) directly downwind from the accident. The risk to the affected population is a measure of the radiological risk posed to society as a whole by the alternative being considered. The impact to the affected population is used as the primary means of comparing various alternatives.

The following provides a summary of transportation impacts. Appendix G details the methods and assumptions used.

4.9.1 Onsite Transportation Impacts - No Action Alternative

Under all alternatives, EBR-II driver spent nuclear fuel would be shipped by DOE in 17 shipments using the TN-FSU cask, or 43 shipments using the NAC-LWT cask. The analysis assumes that 43 shipments are made. Fifteen ceramic waste form, 1 metallic waste form, and 355 spent nuclear fuel shipments would be made from ANL-W to the INEEL Dry Transfer Facility. The total distance traveled on public roads on the INEEL site by trucks carrying radioactive materials would be 16,000 kilometers (9,900 miles).

Impacts of Onsite Incident-Free Transportation. The dose to transportation workers from all transportation activities required by this alternative was estimated at 0.003 person-rem; the collective dose to the affected population would be 0.022 person-rem. Accordingly, incident-free transportation of radioactive material would result in 1.2×10^{-6} latent fatal cancers among transportation workers and 0.000011 latent fatal cancers in the total affected population over the duration of the transportation activities. Latent fatal cancers associated with radiological releases were estimated by multiplying the worker dose by 0.0004 latent fatal cancers per person-rem of exposure, and multiplying the collective dose to the affected population by 0.0005 latent fatal cancers per person-rem of exposure (ICRP 1991).

Impacts of Onsite Accidents During Ground Transportation. The maximum foreseeable onsite transportation accident under this alternative (probability of occurrence: more than 1×10^{-7} per year) would not breech the transportation cask. The probability of more severe accidents also was evaluated, and was estimated to be less than 1×10^{-7} per year.

Estimates of the total ground transportation accident risks under this alternative are as follows: a collective dose to the affected population of less than 1×10^{-6} person-rem would result in less than 1×10^{-9} latent fatal cancers; and traffic accidents would result in 0.00012 traffic fatalities.

4.9.2 Onsite Transportation Impacts - Alternative 1

In addition to the 43 shipments of EBR-II driver spent nuclear fuel, 125 ceramic waste form and 5 metallic waste form shipments from ANL-W to the INEEL Dry Transfer Facility would be made. The total distance traveled on public roads on the INEEL site by trucks carrying radioactive materials would be 6,700 kilometers (4,200 miles).

Impacts of Onsite Incident-Free Transportation. The dose to transportation workers from all transportation activities entailed by this alternative was estimated at 0.0044 person-rem; the collective dose to the affected population would be 0.033 person-rem. Accordingly, incident-free transportation of radioactive material would result in 1.8×10^{-6} latent fatal cancers among transportation workers and 0.000016 latent fatal cancers in the total affected population over the duration of the transportation activities.

Impacts of Onsite Accidents During Ground Transportation. The maximum foreseeable onsite transportation accident under this alternative (probability of occurrence: more than 1×10^{-7} per year) would not breech the transportation cask. The probability of more severe accidents also was evaluated, and was estimated to be less than 1×10^{-7} per year.

Estimates of the total ground transportation accident risks under this alternative are as follows: a collective dose to the affected population of less than 1×10^{-6} person-rem would result in less than 1×10^{-9} latent fatal cancers; and traffic accidents would result in 0.000052 traffic fatalities.

4.9.3 Onsite Transportation Impacts - Alternative 2

In addition to the 43 shipments of EBR-II driver spent nuclear fuel, 27 ceramic waste form, 7 metallic waste form, and 63 spent nuclear fuel shipments from ANL-W to the INEEL Dry Transfer Facility would be made. The total distance traveled on public roads on the INEEL site by trucks carrying radioactive materials would be 5,200 kilometers (3,200 miles).

Impacts of Onsite Incident-Free Transportation. The dose to transportation workers from all transportation activities entailed by this alternative was estimated at 0.0043 person-rem; the collective dose to the affected population would be 0.032 person-rem. Accordingly, incident-free transportation of radioactive material would result in 1.7×10^{-6} latent fatal cancers among transportation workers and 0.000016 latent fatal cancers in the total affected population over the duration of the transportation activities.

Impacts of Onsite Accidents During Ground Transportation. The maximum foreseeable onsite transportation accident under this alternative (probability of occurrence: more than 1×10^{-7} per year) would not breech the transportation cask. The probability of more severe accidents also was evaluated, and was estimated to be less than 1×10^{-7} per year.

Estimates of the total ground transportation accident risks under this alternative are as follows: a collective dose to the affected population of less than 1×10^{-6} person-rem would result in less than 1×10^{-9} latent fatal cancers; and traffic accidents would result in 0.00008 traffic fatalities.

4.9.4 On- and Offsite Transportation Impacts - Alternative 3

In addition to the 43 shipments of EBR-II driver spent nuclear fuel, Alternative 3 would require 11 shipments of declad EBR-II blanket material and 18 shipments of Fermi-1 blanket material from ANL-W to SRS. Twenty-seven ceramic waste form and 2 metallic waste form shipments from ANL-W to the INEEL Dry Transfer Facility would be made. The impacts for these alternatives include both on- and offsite transportation. The total distance traveled on public roads by trucks carrying radioactive materials would be 111,800 kilometers (69,500 miles).

Impacts of On- and Offsite Incident-Free Transportation. The dose to transportation workers from all transportation activities required by this alternative was estimated at 0.0052 person-rem; the collective dose to the affected population would be 0.042 person-rem. Accordingly, incident-free transportation of radioactive material would result in 2.1×10^{-6} latent fatal cancers among transportation workers and 0.000021 latent fatal cancers in the total affected population over the duration of the transportation activities. The dose to transportation workers from transporting cleaned and declad blanket spent nuclear fuel to SRS was estimated at 0.0012 person-rem; the collective dose to the affected population would be 0.012 person-rem. Accordingly, incident-free transportation of radioactive material would result in 4.7×10^{-7} latent fatal cancers among transportation workers and 6×10^{-6} latent fatal cancers in the total affected population over the duration of the transportation activities. The estimated number of nonradiological fatalities from vehicular emissions associated with this alternative is 0.00039.

Impacts of On- and Offsite Accidents During Ground Transportation. The maximum foreseeable offsite transportation accident under this alternative (probability of occurrence: more than 1×10^{-7} per year) would be shipment of EBR-II blanket material from DOE's facility at ANL-W to SRS with a Severity Category 5 accident in a suburban population zone under neutral (average) weather conditions. The accident could result in a dose of 0.00024 person-rem to the public with an associated 1.2×10^{-7} latent fatal cancers, and 2.5×10^{-13} rem to the hypothetical maximally exposed individual with a latent fatal cancer risk of 1.3×10^{-15}. No fatalities would be expected. The probabilities of more severe accidents, different weather conditions at the time of accident, or occurrence in a more densely populated area also were evaluated, and were estimated to be less than 1×10^{-7} per year.

Estimates of the total ground transportation accident risks under this alternative are as follows: a collective dose to the affected population of 3.4×10^{-6} person-rem would result in 1.7×10^{-9} latent fatal cancers; and traffic accidents would result in 0.0018 traffic fatalities. Ground transportation accident risks to the affected population from transporting blanket fuel to SRS were estimated at 3×10^{-6} person-rem, resulting in 1.5×10^{-9} latent fatal cancers; and traffic accidents would result in 0.0017 traffic fatalities.

4.9.5 Onsite Transportation Impacts - Alternative 4

In addition to the 43 shipments of EBR-II driver spent nuclear fuel, 27 ceramic waste form, 2 metallic waste form, and 114 melt and dilute product shipments would be made from ANL-W to the INEEL Dry Transfer Facility. The total distance traveled on public roads on the INEEL site by trucks carrying radioactive materials would be 7,200 kilometers (4,500 miles).

Impacts of Onsite Incident-Free Transportation. The dose to transportation workers from all transportation activities required by this alternative was estimated at 0.02 person-rem; the collective dose to the public would be 0.14 person-rem. Accordingly, incident-free transportation of radioactive material would result in 7.9×10^{-6} latent fatal cancers among transportation workers, and 0.000072 latent fatal cancers in the total affected population over the duration of the transportation activities.

Impacts of Onsite Accidents During Ground Transportation. The maximum foreseeable onsite transportation accident under this alternative (probability of occurrence: more than 1×10^{-7} per year) would not breech the transportation cask. The probability of more severe accidents also was evaluated, and estimated to be less than 1×10^{-7} per year.

Estimates of the total ground transportation accident risks under this alternative are as follows: a collective dose to the affected population of less than 1×10^{-6} person-rem would result in less than 1×10^{-9} latent fatal cancers; and traffic accidents would result in 0.00011 traffic fatalities.

4.9.6 On- and Offsite Transportation Impacts - Alternative 5

In addition to the 43 shipments of EBR-II driver spent nuclear fuel, Alternative 5 requires 11 shipments of declad EBR-II blanket material and 18 shipments of Fermi-1 blanket material from ANL-W to SRS. Twenty-seven ceramic waste form and 2 metallic waste form shipments would be made from ANL-W to the INEEL Dry Transfer Facility. The impacts for these alternatives include both on- and offsite transportation. The total distance traveled on public roads by trucks carrying radioactive materials would be 111,800 kilometers (69,500 miles).

Impacts of On- and Offsite Incident-Free Transportation. The dose to transportation workers from all transportation activities required by this alternative was estimated at 0.0052 person-rem; the collective dose to the public would be 0.042 person-rem. Accordingly, incident-free transportation of radioactive material would result in 2.1×10^{-6} latent fatal cancers among transportation workers and 0.000021 latent fatal cancers in the total affected population over the duration of the transportation activities. The dose to transportation workers from transporting cleaned and declad blanket spent nuclear fuel to SRS was estimated at 0.0012 person-rem; the collective dose to the public would be 0.012 person-rem. Accordingly, incident-free transportation of radioactive material would result in 4.7×10^{-7} latent fatal cancers among transportation workers and 6×10^{-6} latent fatal cancers in the total affected population over the duration of the transportation activities. The estimated number of nonradiological fatalities from vehicular emissions associated with this alternative would be 0.00039.

Impacts of On- and Offsite Accidents During Ground Transportation. The maximum foreseeable offsite transportation accident under this alternative (probability of occurrence: more than 1×10^{-7} per year) would be shipment of EBR-II blanket material from DOE's facility at ANL-W to SRS with a Severity Category 5 accident in a suburban population zone under neutral (average) weather conditions. The accident could result in a dose of 0.00024 person-rem to the public with an associated 1.2×10^{-7} latent fatal cancers, and 2.5×10^{-13} rem to the hypothetical maximally exposed individual with a latent fatal cancer risk of 1.3×10^{-15}. No fatalities would be expected to occur. The probabilities of more severe accidents, different weather conditions at the time of accident, or occurrence in a more densely populated area also were evaluated, and were estimated to be less than 1×10^{-7} per year.

Estimates of the total ground transportation accident risks under this alternative are as follows: a collective dose to the affected population of 3.4×10^{-6} person-rem would result in 1.7×10^{-9} latent fatal cancers; and traffic accidents would result in 0.0018 traffic fatalities. Ground transportation accident risks to the affected population from transporting blanket fuel to SRS were estimated at 3×10^{-6} person-rem, resulting in 1.5×10^{-9} latent fatal cancers; and traffic accidents would result in 0.0017 traffic fatalities.

4.9.7 Onsite Transportation Impacts - Alternative 6

In addition to the 43 shipments of EBR-II driver spent nuclear fuel, 32 ceramic waste form, 1 metallic waste form, and 164 melt and dilute product shipments would be made from ANL-W to the INEEL Dry Transfer

Facility. The total distance traveled on public roads on the INEEL site by trucks carrying radioactive materials would be 9,300 kilometers (5,800 miles).

Impacts of Onsite Incident-Free Transportation. The dose to transportation workers from all transportation activities required by this alternative was estimated at 0.027 person-rem; the collective dose to the public would be 0.20 person-rem. Accordingly, incident-free transportation of radioactive material would result in 0.000011 latent fatal cancers among transportation workers and 0.0001 latent fatal cancers in the total affected population over the duration of the transportation activities.

Impacts of Onsite Accidents During Ground Transportation. The maximum foreseeable onsite transportation accident under this alternative (probability of occurrence: more than 1×10^{-7} per year) would not breech the transportation cask. The probability of more severe accidents also was evaluated, and was estimated to be less than 1×10^{-7} per year.

Estimates of the total ground transportation accident risks under this alternative are as follows: a collective dose to the affected population of less than 1×10^{-6} person-rem would result in less than 1×10^{-9} latent fatal cancers; and traffic accidents would result in 0.00014 traffic fatalities.

4.10 PREFERRED ALTERNATIVE

DOE has identified electrometallurgical treatment (Alternative 1) as its Preferred Alternative for the treatment and management of all sodium-bonded spent nuclear fuel, except for the Fermi-1 blanket fuel. The No Action Alternative is preferred for the Fermi-1 blanket spent nuclear fuel; therefore, Fermi-1 blanket spent nuclear fuel would remain in storage, pending a subsequent decision on its long-term management. While EBR-II spent nuclear fuel is undergoing electrometallurgical treatment, DOE has approximately four years[1] in which to evaluate the operating experience of electrometallurgical treatment technology and continue to investigate alternative treatment techniques that currently require additional development for the Fermi-1 blanket spent nuclear fuel. After this data is evaluated, DOE would decide whether to treat the Fermi-1 blanket spent nuclear fuel using electrometallurgical treatment or to use another treatment method and/or disposal technique.

The environmental impacts of the Preferred Alternative, as identified above, are provided in detail in Section 4.2 for the No Action Alternative and in Section 4.3 for Alternative 1. The evaluations provided in these sections cover treatment of both Fermi-1 blanket and other sodium-bonded spent nuclear fuel. The environmental impact contributions from treatment of the Fermi-1 blanket spent nuclear fuel as compared to the EBR-II sodium-bonded spent nuclear are negligible for all resources except for waste management. Overall, the environmental impacts of Alternative 1 bound those of the Preferred Alternative for all resources except for waste management, where the No Action Alternative bounds. The decision to electrometallurgically treat all sodium-bonded spent nuclear fuel except the Fermi-1 blanket fuel would reduce the treatment duration under Alternative 1 from 13 to 7 years. Storing Fermi-1 blanket spent nuclear fuel pending a subsequent decision on its long-term management would not change the duration of the No Action Alternative, i.e., it would remain 35 years.

Should DOE decide to treat Fermi-1 blanket fuel using a treatment method or process, other than electrometallurgical treatment, that was analyzed in Sections 4.4 through 4.7 of this EIS, the environmental consequences would be equal to or bounded by the EIS. As indicated in these sections, all the alternatives analyzed would result in very small and essentially indistinguishable impacts to public and occupational health and safety, air quality, water resources, environmental justice, and transportation. The volumes of waste generated by separate treatment of Fermi-1 blanket fuel would be equal to or bounded by the values

[1] *Even though it would take six years to electrometallurgically treat EBR-II spent nuclear fuel at ANL-W, for planning purposes, DOE would need to make the decision in four years.*

presented for each of the alternatives analyzed in detail. A decision by DOE to treat some or all of the sodium-bonded blanket fuel using a method which has not been analyzed in detail in this EIS would require an evaluation of associated environmental impacts under a separate NEPA document.

4.11 CUMULATIVE IMPACTS

The Council on Environmental Quality regulations implementing NEPA procedural provisions define cumulative impacts as the impacts on the environment which result from the incremental impact of the action when added to other past, present, and reasonably foreseeable future actions, regardless of what agency (Federal or nonfederal) or person undertakes such other actions (40 CFR 1508.7). The cumulative impacts analysis presented in this section is based on the incremental contribution from the maximum impacts from the proposed action added to the baseline conditions at ANL-W and SRS, as well as the maximum impacts from other on- and offsite past, present, and other reasonably foreseeable future actions. Although it is unlikely that the alternative with the maximum impacts would be implemented to treat and manage sodium-bonded spent nuclear fuel at ANL-W and SRS, it was used to estimate cumulative impacts to ensure a conservative analysis. In accordance with a handbook recently prepared by the Council on Environmental Quality, DOE identified the resource areas in which the treatment and management of sodium-bonded spent nuclear fuel could add to the impacts of past, present, and reasonably foreseeable actions within the project impact zones, as defined by the Council on Environmental Quality (CEQ 1997).

Based on an examination of the environmental impacts of the proposed action, coupled with DOE and other agency actions, it was determined that cumulative impacts for the following areas need to be presented: (1) air resources, (2) water resources, (3) socioeconomics, (4) public and worker health, (5) environmental justice, and (6) waste generation. Discussions of cumulative impacts for land resources, site infrastructure, geologic resources, ecological resources, and cultural and paleontological resources were omitted because the related impacts from the proposed treatment and management of sodium-bonded spent nuclear fuel would either not occur or be so small that their potential contribution to cumulative impacts would be negligible.

For determining the impacts to air, water, socioeconomic, human health, environmental justice, and waste generation resources from commercial and Federal nuclear facilities, the 80-kilometer (50-mile) radius surrounding ANL-W and SRS was selected as the project impact zone. For liquid releases from SRS, the downstream population that uses the Savannah River as its source of drinking water was included in the SRS project impact zone.

Cumulative transportation impacts are discussed at the end of this section.

4.11.1 ANL-W and INEEL

Significant offsite activities within a 80-kilometer (50-mile) radius of ANL-W and INEEL that potentially would contribute to the cumulative environmental impacts presented in this analysis include the System Integration Corporation quartzite mining operation in Arco Hills and the Food, Machinery, and Chemical Corporation, a phosphate processing operations in Pocatello, Idaho. The Food, Machinery, and Chemical Corporation is a primary source for offsite radiological emissions. These emissions have been evaluated by the EPA. Radiological impacts from the operation of the phosphate processing operations are minimal, and are not included in assessments at INEEL (DOE 1999a).

The counties surrounding ANL-W and INEEL have a number of existing and planned industrial and commercial facilities with permitted air emissions and water usage. Because of the distances between ANL-W and INEEL and the private industrial facilities, there is little opportunity for the interaction of plant emissions and no significant contribution to the cumulative impact on air or water. Reasonably foreseeable offsite actions evaluated in this EIS are presented in **Table 4–64**.

Table 4–64 Offsite Activities Included in the Assessment of Cumulative Impacts at ANL-W and INEEL

Activity	*Description*
Housing development, Idaho Falls	300-unit single family housing development planned on approximately 150 acres of vacant land.
Business park, Rexburg	50 acres of vacant land between two light industrial facilities planned for expansion into a light industrial/business park for 30-40 businesses.
Manufacturer, Pocatello	Existing manufactured-home factory to expand from approximately 50 to between 140 and 150 employees. Expansion of 22 acres in Pocatello Airport Industrial Park.
Food, Machinery, and Chemical Corp., Pocatello	Phosphate manufacturing plant to reduce number of furnaces from four to three within the next two years; 25-30 jobs could be lost.
System Integration Corporation Arco Hills Quartzite Mine	Quartzite mining operation and ore processing near Arco Hills on 56 acres. Fourteen acres would be disturbed by the quarry operation and a small waste ore dump; 22 acres would be disturbed by the construction of a haul road; 11 acres would be disturbed by the ore crushing facilities; and 9 acres would be disturbed by the loading facilities at INEEL. The project would employ 40 workers.

Source: DOE 1999a.

The cumulative impacts analysis also addressed the contributory effects from other past, present, and reasonably foreseeable future DOE actions at ANL-W and INEEL. These actions and their associated NEPA documentation are summarized in Section 1.6. The contributory effects of impacts from actions proposed in the *Idaho High-Level Waste and Facilities Disposition Environmental Impact Statement* (DOE 1999e) were included in the cumulative impact analysis. In this EIS, DOE evaluated the impacts from the proposed construction and operation of a high-level radioactive waste and liquid sodium-bearing waste treatment facility at INEEL to make these materials ready for disposal. This project also involves the disposition of high-level radioactive waste generation, storage, and treatment facilities at INEEL upon the completion of their missions.

Other reasonably foreseeable future actions that may contribute to cumulative impacts at INEEL but were not included in this analysis include a proposed DOE Office of Nuclear Energy, Science, and Technology project. This project involves evaluating INEEL as a potential site for the production of plutonium-238 for use in radioisotope power systems for future space missions. This project would include the use of INEEL's Fluorinel Dissolution Process Facility at INTEC for either storing neptunium-237 and/or fabricating and processing neptunium-237 targets to produce plutonium-238, and the use of the Advanced Test Reactor for the irradiation of neptunium-237 targets. The Advanced Test Reactor is an operating test reactor with a programmatic mission to support the Naval Reactor Fuels Program. Public scoping for this project has been completed. A preliminary review of the project indicates that there would be a contributory effect to the cumulative impacts—primarily to public and worker health and safety due to the fabrication and processing of neptunium-237 targets in the Fluorinel Dissolution Process Facility, loading and unloading targets in the Advanced Test Reactor, and handling of irradiated targets for packaging and shipping. The cumulative impacts from this proposed project will be addressed in a separate NEPA document.

The proposed commercial project (VentureStar) would involve a commercial spin-off of the National Aeronautics and Space Administration's Reusable Launch Vehicle research program that would replace the existing Space Shuttle Program. INEEL is being considered as a potential candidate site for both the launch and landing of this next-generation spacecraft. The project is in the very early stages of development and does not appear to be near term (5 to 10 years). Cumulative impacts from this proposed project would be addressed in separate NEPA documentation.

The cumulative impacts analysis also included the impacts from actions proposed in this EIS. Risks to members of the public and site workers from radiological and nonradiological releases were based on operational impacts from the alternatives described in Chapter 4 of this EIS.

Temporal limits were defined by examining the period of influence from both the proposed action and other Federal and nonfederal actions that have the potential for cumulative impacts. Actions to support the treatment of sodium-bonded blanket spent nuclear fuel at ANL-W are expected to begin in 2000 in preparation for ultimate offsite disposal, possibly in a geologic repository which probably will not be available until at least 2010. Final offsite shipments of spent nuclear fuel at ANL-W and INEEL for disposal would be completed by 2035.

The period of interest for the cumulative impacts analysis for this EIS includes the proposed construction, operation, and disposition of facilities identified in the *Idaho High-Level Waste and Facilities Disposition Environmental Impact Statement* (DOE 1999e) and the *Advanced Mixed Waste Treatment Project Final Environmental Impact Statement* (DOE 1999a), while actions for other nuclear materials and surplus plutonium disposition would be ongoing.

4.11.1.1Air Resources

Table 4–65 compares the cumulative concentrations of nonradiological air pollutants from INEEL to Federal and state regulatory standards. The listed values are the maximum modeled concentrations that could occur at the ground level at the site boundary. The data demonstrate that the total estimated concentrations of nonradiological air pollutants from INEEL in the all cases would be well below the regulatory standards at the site boundary. Among the pollutants, the concentrations of nitrogen oxides come closest to the standard (14 percent of the standard for the annual averaging time). The remaining pollutant emissions would result in concentrations below 13 percent of the applicable standards. As indicated in this table, the values presented in the INEEL baseline include concentrations from releases at ANL-W. ANL-W's criteria pollutant concentrations are from currently operating equipment, which are not expected to increase under any of the alternatives. Therefore, there would be no contribution to cumulative air quality impacts at INEEL as a result of the proposed action.

4.11.1.2Water Resources

There would be no liquid effluent released to surface water or groundwater from the operation of ANL-W or INEEL facilities as a result of the proposed action. Therefore, there would be no contribution to the cumulative impact.

4.11.1.3Socioeconomic Impacts

No additional workers would be required for the operation of ANL-W or INEEL facilities as a result of the proposed action. Therefore, there would be no contribution to the cumulative impact.

4.11.1.4Public and Worker Health

Table 4–66 summarizes the cumulative radiological health effects of routine ANL-W and INEEL operations, proposed DOE actions, and nonfederal nuclear facility operations. Impacts resulting from proposed DOE actions are described in the various EISs listed in Section 1.6. In addition to estimated radiological doses to the maximally exposed offsite individual, the offsite population, and workers, Table 4–66 lists the potential number of latent cancer fatalities for the public and workers from exposure to radiation. The radiation dose to the maximally exposed offsite individual would be 0.047 millirem per year, which is well below the applicable DOE regulatory limits (10 millirem per year from the air pathway [40 CFR 61] and

100 millirem per year for all pathways). The total annual population dose of 0.35 person-rem for current and projected activities translates into 0.00017 latent cancer fatalities for each year of exposure for the population living within a 80-kilometer (50-mile) radius of the ANL-W.

Table 4–65 Estimated Maximum Cumulative Ground-Level Concentrations of Nonradiological Pollutants (micrograms per cubic meter) at the INEEL Boundary

Pollutant	*Averaging Time*	*Most Stringent Standard or Guideline* [a]	*INEEL Baseline* [b]	*Advanced Mixed Waste Treatment Project EIS* [c]	*Idaho High-Level Waste and Facilities Disposition EIS* [d]	*Cumulative Concentrations* [e]
Carbon monoxide	8 hours	10,000 [f]	120	1	4.2	130
	1 hour	40,000 [f]	265	115	10	390
Nitrogen oxides	Annual	100 [f]	13	0.3	0.2	14
PM_{10}	Annual	50 [f]	0.65	0.006	0.02	1
	24 hours	150 [f]	13	4.6	0.3	18
	24 hours (99^{th} percentile over 3 years)	150 [g]	Not available	Not available	Not available	Not available
$PM_{2.5}$	3-year annual	15 [g]	Not available	Not available	Not available	Not available
	24 hours (98^{th} percentile over 3 years)	65 [g]	Not available	Not available	Not available	Not available
Sulfur dioxide	Annual	80 [f]	3.4	0.012	0.57	4
	24 hours	365 [f]	32	4.5	9	46
	3 hours	1,300 [f]	84	25	42	151

PM_n = Particulate matter less than or equal to *n* microns in diameter.

[a] The more stringent Federal or state standard is presented if both exist for the averaging period.

[b] INEEL baseline includes concentrations from releases at ANL-W which, in turn, include releases under all alternatives considered in the SBSNF EIS (see Section 3.2.3).

[c] DOE 1999a: Table 5.7-6, Preferred Alternative (Microencapsulation option).

[d] DOE 1999e: Table C.2-14, Separation (Planning Basis) option.

[e] Values presented in this column could be different from the sum of the individual values due to rounding.

[f] Federal and state standard.

[g] Federal standard.

The annual collective dose to the worker population would be 200 person-rem. In addition, doses to individual workers would be kept below the DOE Administrative Control Level of 2,000 millirem per year, as established for all DOE activities in DOE Order N 441.1, which is well below the regulatory limit of 5,000 millirem per year (10 CFR 835). Furthermore, "as low as reasonably achievable" principles would be exercised to maintain individual worker doses below the DOE Administrative Control Level of 2,000 millirem per year.

4.11.1.5Environmental Justice

As discussed in Chapter 4 and Appendix H, implementation of the alternatives for the treatment and management of sodium-bonded spent nuclear fuel at ANL-W and INEEL would have no significant impact on public health or the environment. Therefore, the implementation of the proposed action or the No Action Alternative would result in no disproportionately high and adverse impacts on minority or low-income populations residing within potentially affected areas.

Table 4–66 Estimated Annual Cumulative Radiological Doses and Resulting Health Effects to Offsite Population and Facility Workers at ANL-W and INEEL

	Maximally Exposed Offsite Individual		*Population* [a]		*Workers*	
Activity	*Dose (Millirem)*	*Latent Cancer Fatality Risk*	*Collective Dose (Person-Rem)*	*Excess Latent Cancer Fatalities*	*Collective Dose (Person-Rem)*	*Excess Latent Cancer Fatalities*
ANL-W and INEEL Baseline [b]	0.021	1.1×10^{-8}	0.23	0.00012	115	0.046
SBSNF EIS [c]	0.002	1.0×10^{-9}	0.012	6.0×10^{-6}	22	0.0088
Advanced Mixed Waste Treatment Program [d]	0.022	1.1×10^{-8}	0.009	4.5×10^{-6}	4.1 [e]	0.0016
High-Level Waste and Facilities Disposition [f]	0.002	1.0×10^{-9}	0.10	0.00005	59	0.024
Total	0.047	2.4×10^{-8}	0.35	0.0017	200	0.08

[a] A collective dose to the 80-kilometer (50-mile) population from atmospheric releases. There would be no liquid releases from ANL-W and INEEL facilities as a result of the proposed action.
[b] Data from Tables 3–9 and 3–10 of this SBSNF EIS.
[c] Alternative 6: Melt and dilute blanket and driver fuel at ANL-W.
[d] DOE 1999a: Tables 5.12–1 and E.4–7. Preferred Alternative (Microencapsulation Option) Record of Decision (64 FR 16948).
[e] Estimate based on the number of workers and the average dose per worker, i.e., 50 workers (DOE 1999a: Table E.4–7) × 81 millirem (DOE 1999a: Table 5.12–1) = 4050 person millirem = 4.1 person-rem.
[f] DOE 1999e: Table 5.4–6, maximum dose for any alternative. Average annual dose of 190 millirem per worker.

4.11.1.6Waste Generation

As stated in the Waste Management discussions for each alternative presented earlier in Chapter 4, low-level radioactive waste, mixed and hazardous waste, and transuranic waste would be generated by the treatment of sodium-bonded spent nuclear fuel. Under the proposed action (with the partial exception of Alternative 2), the existing sodium-bonded spent nuclear fuel inventories at ANL-W and INEEL would be converted into a high-level radioactive waste form for disposal in a geologic repository and, therefore, the volume of the high-level radioactive waste that would be generated is not counted as new waste—"high-level radioactive waste." In fact, under the proposed action, the amount of material at ANL-W and INEEL scheduled for disposal in a geologic repository would decrease. For all alternatives under the proposed action, the volume of the new high-level radioactive waste forms would be less than the volume of untreated sodium-bonded spent nuclear fuel (the No Action Alternative). However, as stated in the Waste Management discussions, the projected amount of high-level radioactive waste would not require additional treatment and storage capacities beyond the current and planned INEEL capacities.

Table 4–67 lists the cumulative total waste generated at ANL-W and INEEL for years 2000 to 2035. The estimated quantity of radioactive/hazardous waste from baseline operations in this forecast through the year 2035 would be 205,550 cubic meters (7.25 million cubic feet). Waste generated by Alternative 6: Melt and dilute blanket and driver fuel at ANL-W (the alternative generating the most waste in all categories) would add a total of 980 cubic meters (34,610 cubic feet). During a 15-year time period, other reasonably foreseeable activities associated with the treatment of high-level radioactive waste and facility disposition at INEEL could add an additional 30,730 cubic meters (1.1 million cubic feet). Therefore, the potential cumulative total amount of waste generated from ANL-W and INEEL activities would be 237,260 cubic meters (8.4 million cubic feet).

Table 4–67 Estimated Cumulative Total Waste Generation for Years 2000 to 2035 From ANL-W and INEEL Concurrent Activities (Cubic Meters)

Waste Type	*ANL-W and INEEL Baseline Operations* [a]	*Idaho HLW and Facility Disposition EIS* [a]	*SBSNF EIS* [b]	*Total*
High-level radioactive	0	0	0 [d]	0 [c]
Low-level radioactive	135,600	15,320	925	151,845
Hazardous/mixed low-level radioactive	4,950	15,300	40	20,290
Transuranic	65,000[d]	110	15	65,125
Total	205,550	30,730	980	237,260

HLW = High-level radioactive waste.

[a] DOE 1999e: Figures 5.4–1 through 5.4–3 and input values for those figures through year 2035, Separations Alternative. Maximum quantities for any alternative.

[b] Alternative 6: Melt and dilute blanket and driver fuel at ANL-W; 12 years of operation. This alternative would generate the most waste in all categories.

[c] During treatment, the sodium-bonded spent nuclear fuel from existing inventories at ANL-W and INEEL would be converted into a high-level radioactive waste form for disposal in a geologic repository. For any alternative, the amount of material at ANL-W and INEEL scheduled for disposal in a geologic repository would not increase.

[d] In storage at the Radioactive Waste Management Complex.

The Central Facilities Area and Bonneville County landfill accepts nonhazardous and nonradioactive solid waste generated at INEEL. The onsite landfill complex was designed to accommodate combined ANL-W and INEEL solid waste disposal needs for a projected maximum operational life of 30 years.

The activities supporting the treatment and management of sodium-bonded spent nuclear fuel and other planned ANL-W and INEEL activities would not generate larger volumes of radioactive, hazardous, or solid waste beyond the current and projected capacities of ANL-W and INEEL waste storage and/or management facilities.

4.11.2 Savannah River Site

Nuclear facilities within a 80-kilometer (50-mile) radius of SRS include Georgia Power's Vogtle Electric Generating Plant across the river from SRS; Chem-Nuclear Inc., a commercial low-level radioactive waste burial site just east of SRS; and Starmet CMI, Inc. (formerly Carolina Metals), located southeast of SRS, which processes uranium-contaminated metals. Radiological impacts from the operation of the Vogtle Electric Generating Plant, a two-unit commercial nuclear power plant, are minimal, but DOE has factored them into the analysis. As stated in the SRS Spent Nuclear Fuel Management Final EIS (DOE 2000), the South Carolina Department of Health and Environmental Control Annual Report indicates that operation of the Chem-Nuclear Services facility and the Starmet CMI facility do not noticeably impact radiation levels in air or liquid pathways in the vicinity of SRS. Therefore, they are not included in this assessment.

The counties surrounding SRS have numerous existing (e.g., textile mills, paper product mills, and manufacturing facilities) and planned (e.g., Bridgestone Tire) industrial facilities with permitted air emissions and discharges to surface waters. Because of the distances between SRS and the private industrial facilities, there is little opportunity for interactions of plant emissions, and no major cumulative impact on air or water quality. Construction and operation of Bridgestone Tire and Hankook Polyester facilities could affect the regional socioeconomic cumulative impacts.

Additional offsite facilities with the potential to affect the nonradiological environment include South Carolina Electric and Gas Company's Urquhart Station. Urquhart Station is a three-unit, 250-megawatt, coal- and natural gas-fired steam electric plant on Beech Island, South Carolina, located north of SRS. Because

of the distance between SRS and the Urquhart Station and the regional wind direction frequencies, there is little opportunity for any interaction of plant emissions, and no significant cumulative impact on air quality.

DOE also evaluated the impacts from its own proposed future actions by examining impacts to resources and the human environment, as shown in NEPA documentation related to SRS (see Section 1.6). Additional NEPA documents related to SRS that are considered in the cumulative impacts section include the following:

Environmental Assessment for the Tritium Facility Modernization and Consolidation Project at the Savannah River Site (DOE 1998a). This environmental assessment addresses the impacts of consolidating the tritium activities currently performed in Building 232-H into the new Building 233-H and Building 234-H. Tritium extraction functions would be transferred to the Tritium Extraction Facility. The overall impact would be to reduce the tritium facility complex net tritium emissions by up to 50 percent. Another positive effect of this planned action would be to reduce the amount of low-level radioactive job-control waste. Effects on other resources would be negligible. Therefore, impacts from the environmental assessment have not been included in this cumulative impacts analysis.

Final Environmental Impact Statement on Management of Certain Plutonium Residues and Scrub Alloy Stored at the Rocky Flats Environmental Technology Site (DOE 1998d). DOE proposes to process certain plutonium-bearing materials being stored at the Rocky Flats Environmental Technology Site. These materials are plutonium residues and scrub alloy remaining from nuclear weapons manufacturing operations formerly conducted by DOE at Rocky Flats. DOE has decided to remove the plutonium from certain residues that would be shipped from the Rocky Flats Environmental Technology Site to SRS for stabilization. The separated plutonium would be stored at SRS pending disposition decisions. Environmental impacts from using the F-Canyon to chemically separate the plutonium from the remaining materials at SRS are included in this section.

Draft and Final Environmental Impact Statements for the Construction and Operation of a Tritium Extraction Facility at the Savannah River Site (DOE 1998b, DOE 1999b). DOE proposes to construct and operate a Tritium Extraction Facility at SRS to provide the capability to extract tritium from commercial light water reactor targets and targets of similar design. The purpose of the proposed action and alternatives evaluated in the EIS is to provide tritium extraction capability to support reactor tritium production. Environmental impacts from the maximum processing option in this EIS are included in this section.

Defense Waste Processing Facility Supplemental Environmental Impact Statement (DOE 1994). The selected alternative in the Record of Decision was the completion and operation of the Defense Waste Processing Facility to immobilize high-level radioactive waste at SRS. The facility is currently processing sludge from SRS high-level radioactive waste tanks. However, SRS baseline data are not representative of full Defense Waste Processing Facility operational impacts, including the processing of salt and supernate from these tanks. Therefore, the Defense Waste Processing Facility data is listed separately.

In addition, the cumulative impacts analysis also includes the impacts from actions proposed in this SBSNF EIS. Risks to members of the public and site workers from radiological and nonradiological releases are based on operational impacts from the alternatives described in Sections 4.5 and 4.7, of this EIS.

Temporal limits were defined by examining the period of influence from both the proposed action and other Federal and nonfederal actions that have the potential for cumulative impacts. Actions to support the treatment of sodium-bonded blanket spent nuclear fuel at SRS are expected to begin in 2003 in preparation for ultimate offsite disposal, possibly in a geologic repository which probably will not be available until at least 2010. Final offsite shipments for spent nuclear fuel currently assigned to SRS for disposal would be completed by 2035.

The period of interest for the cumulative impacts analysis for this EIS includes the proposed construction and operation of facilities identified in the SRS Spent Nuclear Fuel Management Final EIS (DOE 2000) and the Draft EIS for the SRS Tritium Extraction Facility (DOE 1998b, DOE 1999b), while actions for nuclear materials, highly enriched uranium, and surplus plutonium disposition would be ongoing.

4.11.2.1Air Resources

Table 4–68 compares the cumulative concentrations of nonradiological air pollutants from SRS to Federal and state regulatory standards. The listed values are the maximum modeled concentrations that could occur at ground level at the site boundary. The data demonstrate that total estimated concentrations of nonradiological air pollutants from SRS in all cases would be below the regulatory standards at the site boundary. Among the pollutants, the concentration of sulfur dioxide comes closest to the standard (approximately 96 percent of the standard for the 24-hour averaging time). The remaining pollutant emissions would range from 25 to 93 percent of the applicable standards.

Table 4–68 Estimated Maximum Cumulative Ground-Level Concentrations of Nonradiological Criteria Pollutants (Micrograms per Cubic Meter) at the SRS Boundary

Pollutant	*Averaging Time*	*Most Stringent Standard or Guideline* [a]	*SRS Baseline* [b]	*SBSNF EIS* [c]	*Other Foreseeable Planned SRS Activities* [d]	*Cumulative Concentrations* [e]
Carbon monoxide	8 hours	10,000 [f]	6,900	1.22	6.78	6,908
	1 hour	40,000 [f]	10,000	9.06	44.63	10,054
Nitrogen dioxide	Annual	100 [f]	26	3.11	4.63	34
PM_{10}	Annual	50 [f]	25	Less than 0.01	0.21	25
	24 hours (interim)	150 [f]	130	0.11	6.82	137
	24 hours (99^{th} percentile over 3 years)	150 [g]	(h)	Not available	Not available	Not available
$PM_{2.5}$	3-year annual	15 [g]	(h)	Not available	Not available	Not available
	24 hours (98^{th} percentile over 3 years)	65 [g]	(h)	Not available	Not available	Not available
Sulfur dioxide	Annual	80 [f]	34	Less than 0.01	0.06	34
	24 hours	365 [f]	350	0.12	0.96	351
	3 hours	1,300 [f]	1,200	0.91	5.28	1,206

PM_n = Particulate matter less than or equal to *n* microns in diameter.
[a] The more stringent Federal or state standard is presented if both exist for the averaging period.
[b] Data from Table 3–16 of this EIS.
[c] Alternative 3: PUREX process blanket fuel at SRS F-Canyon.
[d] Data compiled from SRS Spent Nuclear Fuel Management Final EIS (DOE 2000: Table 5–1), including contributions from the Preferred Alternative less contributions from SBSNF EIS.
[e] Values in this column are rounded to the nearest number.
[f] Federal and state standard.
[g] Federal standard.
[h] No data available with which to assess particulate matter concentrations.

DOE also evaluated the cumulative impacts of airborne radiological releases in terms of dose to a maximally exposed offsite individual at the SRS boundary. DOE included the impacts of the Vogtle Plant (NRC 1996) in this cumulative total. The radiological emissions from the operation of the Chem-Nuclear low-level radioactive waste disposal facility just east of SRS are very low (DOE 2000) and are not included. **Table 4–69** lists the results of this analysis, using 1996 emissions (1992 for the Vogtle Plant) for the SRS baseline. The cumulative dose to the maximally exposed offsite individual would be 0.10 millirem per year, well below the regulatory standard of 10 millirem per year (40 CFR Part 61). Summing the doses to the

maximally exposed offsite individual for the proposed action and baseline SRS operations listed in Table 4–69 is an extremely conservative approach because, to get the calculated dose, the maximally exposed offsite individual would have to occupy different physical locations at the same time, which is impossible.

Table 4–69 Estimated Average Annual Cumulative Radiological Doses and Resulting Health Effects to the Offsite Population in the 80-Kilometer (50-Mile) Radius From Airborne Releases at SRS

Activity	*Maximally Exposed Offsite Individual*		*Population* [a]	
	Dose (millirem)	*Latent Cancer Fatality Risk*	*Collective Dose (person-rem)*	*Excess Latent Cancer Fatalities*
SRS baseline [b]	0.050	2.5×10^{-8}	5.5	0.0028
SBSNF EIS [c]	0.00039	2.0×10^{-10}	0.019	9.5×10^{-6}
Management of spent nuclear fuel [d]	0.015	7.5×10^{-9}	0.56	0.00028
Disposition of surplus highly enriched uranium [d]	0.0025	1.3×10^{-9}	0.16	0.00008
Tritium Extraction Facility [d]	0.02	1.0×10^{-8}	0.77	0.00039
Disposition of surplus plutonium [d]	0.0074	3.7×10^{-9}	1.8	0.0009
Management of plutonium residues/scrub alloy [d]	0.00057	2.9×10^{-10}	0.0062	3.1×10^{-6}
Defense Waste Processing Facility [d]	0.001	5.0×10^{-10}	0.071	0.000036
DOE Complex miscellaneous components [d]	0.0044	2.2×10^{-9}	0.007	3.3×10^{-6}
Vogtle Plant [d]	0.00054	2.7×10^{-10}	0.042	0.000021
Total	0.10	5.1×10^{-8}	8.94	0.0045

[a] A collective dose to the 80-kilometer (50-mile) population.
[b] Data from Table 3–21 of this EIS.
[c] Alternative 3: PUREX process blanket fuel at SRS F-Canyon.
[d] Data from SRS Spent Nuclear Fuel Management Final EIS (DOE 2000: Table 5–2 maximum impact alternative).

Adding the population doses from current and projected activities at SRS, the Vogtle Plant, the SRS Spent Nuclear Fuel Management Final EIS, and this EIS could yield a total annual cumulative dose of 8.94 person-rem from airborne sources. The total annual cumulative dose translates into 0.0045 latent cancer fatalities for each year of exposure for the population living within an 80-kilometer (50-mile) radius of SRS.

4.11.2.2Water Resources

At present, a number of SRS facilities discharge treated wastewater to Upper Three Runs and its tributaries and Fourmile Branch via NPDES–permitted outfalls. These include the F- and H-Area Effluent Treatment Facility and the M-Area Liquid Effluent Treatment Facility. As stated in Sections 4.5.2 and 4.7.2, operations associated with the treatment and management of sodium-bonded spent nuclear fuel are not expected to result in any discharges to groundwater. The only technology that would result in discharges of radiological and nonradiological effluent to surface water would be PUREX processing. The major sources of liquid effluent from facilities associated with PUREX processing would be process cooling water and steam condensate systems that could contain small quantities of radionuclides and chemicals. This process wastewater would be treated at the F-Area Effluent Treatment Facility and then discharged to the Upper Three Runs. Studies of water quality and biota downstream of the Effluent Treatment Facility outfall suggest that discharges have not degraded the water quality of Upper Three Runs (DOE 2000).

Other potential sources of contaminants into Upper Three Runs during the periods in which sodium-bonded spent nuclear fuel would be treated in F-Area using PUREX, or in L-Area using melt and dilute treatment,

include activities described in the SRS Spent Nuclear Fuel Management Final EIS (DOE 2000), the tritium extraction facility, environmental restoration, and decontamination and deactivation activities, as well as modifications to existing SRS facilities. Discharges from activities associated with the SRS Spent Nuclear Fuel Management Final EIS and tritium extraction facility would not add significant amounts of nonradiological contaminants to Upper Three Runs. The amount of discharge associated with environmental restoration and decontamination and deactivation activities would vary based on the level of activity. All the potential activities that could result in wastewater discharges would be required to comply with the NPDES permit limits that ensure protection of water quality.

Table 4–70 summarizes the estimated cumulative radiological doses from waterborne sources to human receptors downstream from SRS. Liquid effluent released to SRS streams that are tributaries of the Savannah River could contain small quantities of radionuclides. The exposure pathways considered in this analysis included drinking water, fish ingestion, shoreline exposure, swimming, and boating. The estimated cumulative dose to the maximally exposed offsite individual from liquid releases would be 0.24 millirem per year, well below the regulatory standard of 4 millirem per year (40 CFR Part 141). Adding the population doses associated with current and projected SRS activities would yield a cumulative annual dose of 2.6 person-rem from liquid sources. This translates into 0.0013 latent cancer fatalities for each year of exposure of the population living within an 80-kilometer (50-mile) radius of SRS.

Table 4–70 Estimated Average Annual Cumulative Radiological Doses and Resulting Health Effects to the Offsite Population in the 80-Kilometer (50-Mile) Radius From Liquid Releases at SRS

	Maximally Exposed Offsite Individual		*Population* [a]	
Activity	*Dose (millirem)*	*Latent Cancer Fatality Risk*	*Collective Dose (person-rem)*	*Excess Latent Cancer Fatalities*
SRS baseline [b]	0.13	6.5×10^{-8}	2.4	0.0012
SBSNF EIS [c]	0.00012	6.0×10^{-11}	0.00068	3.4×10^{-7}
Management of spent nuclear fuel [d]	0.057	2.9×10^{-8}	0.19	0.000095
Disposition of surplus highly enriched uranium [d]	(e)	(e)	(e)	(e)
Tritium Extraction Facility [d]	(e)	(e)	(e)	(e)
Defense Waste Processing Facility [d]	(e)	(e)	(e)	(e)
Disposition of surplus plutonium [d]	(e)	(e)	(e)	(e)
Management plutonium residues/scrub alloy [d]	(e)	(e)	(e)	(e)
DOE Complex miscellaneous components [d]	0.000042	2.1×10^{-11}	0.00024	1.2×10^{-7}
Plant Vogtle [d]	0.054	2.7×10^{-8}	0.0025	1.3×10^{-6}
Total	0.24	1.2×10^{-7}	2.6	0.0013

[a] A collective dose to the 80-kilometer (50-mile) and downstream population.
[b] Data from Table 3–21 of this EIS.
[c] Alternative 3: PUREX process blanket fuel at SRS F-Canyon.
[d] Data from SRS Spent Nuclear Fuel Management Final EIS (DOE 2000: Table 5-3 maximum impact alternative).
[e] Less than minimum reportable levels.

4.11.2.3Socioeconomic Impacts

No additional workers would be required for the operation of PUREX and melt and dilute facilities at SRS as a result of the proposed action. Therefore, there would be no contribution to the cumulative impacts.

4.11.2.4Public and Worker Health

Table 4–71 summarizes the cumulative radiological health effects of routine SRS operations, proposed DOE actions, and nonfederal nuclear facility operations (Vogtle Electric Generating Plant). Impacts resulting from proposed DOE actions are described in the EISs listed previously in this chapter. In addition to estimated radiological doses to the hypothetical maximally exposed offsite individual, the offsite population, and the workers, Table 4–71 lists the potential number of latent cancer fatalities for the public and workers due to radiation exposure. The radiation dose to the maximally exposed offsite individual from air and liquid pathways would be 0.34 millirem per year, which is well below the applicable DOE regulatory limits (10 millirem per year from the air pathway; 4 millirem per year from the liquid pathway; and 100 millirem per year for all pathways). The total annual population dose for current and projected activities of 11.5 person-rem translates into 0.0058 latent cancer fatalities for each year of exposure for the population living within an 80-kilometer (50-mile) radius of SRS.

The annual radiation dose to a worker population would be 859 person-rem. In addition, doses to individual workers would be kept below the DOE Administrative Control Level of 2,000 millirem per year, as established for all DOE activities in DOE Order N 441.1, which is well below the regulatory limit of 5,000 millirem per year (10 CFR 835). Furthermore, standards and practices to ensure worker doses are as low as reasonably achievable would be exercised to maintain individual worker doses below the DOE Administrative Control Level of 2,000 millirem per year.

4.11.2.5Environmental Justice

As discussed in Chapter 4 and Appendix H, implementation of the alternatives for the treatment and management of sodium-bonded spent nuclear fuel at SRS would have no significant impact on public health or the environment. Therefore, the implementation of either of two alternatives at SRS would result in no disproportionately high and adverse impacts on minority or low-income populations residing within potentially affected areas.

4.11.2.6Waste Generation

As stated in Sections 4.5.6 and 4.7.6, high-level and low-level radioactive waste, transuranic waste, and hazardous/mixed waste would be generated from the treatment of declad and cleaned sodium-bonded blanket spent nuclear fuel at SRS. The largest volumes of low-level radioactive and transuranic waste would be generated with PUREX processing. However, as stated in Sections 4.5.6 and 4.7.6, the projected waste generation rates would not require additional treatment and storage capacities beyond the current and planned SRS capacities. It should be noted that the treatment of blanket spent nuclear fuel at SRS would result in the generation of new high-level radioactive waste that would be added to the SRS current inventory. This is because the blanket spent nuclear fuel would be transported from ANL-W to SRS for treatment.

Table 4–72 lists the cumulative volumes of liquid high-level and solid low-level radioactive, transuranic, and hazardous/mixed waste that SRS would generate. The table includes data from the SRS 30-year expected waste forecast. The 30-year expected waste forecast is based on operations, environmental restoration, and decontamination and deactivation waste forecasts from existing generators and the following assumptions: (1) secondary waste from the Defense Waste Processing Facility, In-Tank Precipitation, and Extended Sludge Processing operations are addressed in the Defense Waste Processing Facility EIS; (2) high-level radioactive waste volumes are based on the selected option for the F-Canyon Plutonium Solutions EIS; (3) some investigation-derived waste is handled as hazardous waste per RCRA regulations; (4) purge water from well samplings is handled as hazardous waste; and (5) the continued receipt of small amounts of low-level radioactive waste from other DOE facilities and nuclear naval operations (DOE 2000).

Table 4–71 Estimated Average Annual Cumulative Radiological Doses and Resulting Health Effects to the Offsite Population and Facility Workers at SRS

Activity	*Maximally Exposed Offsite Individual*				*Population* [a]				*Workers*	
	Dose From Airborne Releases (millirem)	*Dose From Liquid Releases (millirem)*	*Total Dose (millirem)*	*Latent Cancer Fatality Risk*	*Dose From Airborne Releases (person-rem)*	*Dose From Liquid Releases (person-rem)*	*Collective Dose (person-rem)*	*Excess Latent Cancer Fatalities*	*Collective Dose (person-rem)*	*Excess Latent Cancer Fatalities*
SRS baseline [b]	0.050	0.13	0.18	9.5×10^{-8}	5.5	2.4	7.9	0.0025	165	0.066
SBSNF EIS [c]	0.00039	0.00012	0.00051	2.6×10^{-10}	0.019	0.00068	0.020	1.0×10^{-8}	38	0.015
Management of spent nuclear fuel [d]	0.015	0.057	0.072	3.6×10^{-8}	0.56	0.19	0.75	0.00038	55	0.022
Disposition of surplus highly enriched uranium [d]	0.0025	(e)	0.0025	1.3×10^{-8}	0.16	(e)	0.16	0.00008	11	0.00044
Tritium Extraction Facility [d]	0.02	(e)	0.02	1.0×10^{-8}	0.77	(e)	0.77	0.00039	4	0.0016
Defense Waste Processing Facility [d]	0.001	(e)	0.001	5.0×10^{-10}	0.071	(e)	0.071	0.000036	120	0.048
Disposition of surplus plutonium [d]	0.0074	(e)	0.0074	3.7×10^{-9}	1.8	(e)	1.8	0.0009	456	0.18
Management plutonium residues/ scrub alloy [d]	0.00057	(e)	0.00057	2.9×10^{-10}	0.0062	(e)	0.0062	3.1×10^{-6}	7.6	0.003
DOE Complex miscellaneous components [d]	0.0044	0.000042	0.0044	2.2×10^{-9}	0.007	0.00024	0.0072	3.6×10^{-6}	2	0.001
Vogtle Plant [d]	0.00054	0.054	0.055	2.7×10^{-8}	0.042	0.0025	0.045	0.000022	Not available	
Total	0.10	0.24	0.34	1.9×10^{-7}	8.95	2.60	11.5	0.0058	859	0.34

[a] A collective dose to the 80-kilometer (50-mile) population for atmospheric releases and to the downstream users of the Savannah River for liquid releases.
[b] Data from Tables 3–21 and 3–22 of this EIS.
[c] Alternative 3: PUREX process blanket fuel at SRS F-Canyon.
[d] Data from SRS Spent Nuclear Fuel Management Final EIS (DOE 2000: Table 5-4).
[e] Less than minimum reportable levels.

Table 4–72 Estimated Cumulative Total Waste Generation From SRS Concurrent Activities (Cubic Meters)

Waste Type	*SRS Baseline Operations* [a]	*SBSNF EIS* [b]	*Spent Nuclear Fuel Management* [a]	*ER/D&D* [a]	*Other Waste Volume* [a]	*Total* [c]
Liquid high-level radioactive	14,129	510	11,000	0	69,040	94,680
Low-level radioactive	118,669	900	140,000	61,630	109,200	430,400
Hazardous/mixed low-level radioactive	3,856	7	270	6,173	4,430	14,740
Transuranic	6,012	90	3,700	0	8,730	18,530
Total [c]	142,670	1,510	154,970	67,800	191,400	558,350

ER/D&D = environmental restoration/decontamination and decommissioning; based on a total 30-year expected waste forecast, including previously generated waste.

[a] Data from SRS Spent Nuclear Fuel Management Final EIS (DOE 2000) maximum impact alternative, Table 5-5, based on a total 30-year expected waste forecast, including previously generated waste, and adjusted for the SBSNF EIS.

[b] Alternative 3: PUREX processing of declad and cleaned blanket spent nuclear fuel at SRS F-Canyon.

[c] The values are rounded to the nearest ten; the total sum may be different from the sum of individuals.

As indicated in Table 4–72, the estimated quantity of radioactive/hazardous waste from SRS operations in this forecast during the next 30 years would be approximately 142,670 cubic meters (5.04 million cubic feet). Waste generated by Alternative 3: PUREX processing of blanket fuel at SRS F-Canyon, would add a total of approximately 1,510 cubic meters (53,330 cubic feet). Waste generated from the conventional (PUREX) processing option described in the SRS Spent Nuclear Fuel Management Final EIS would add a total of 154,970 cubic meters (5.48 million cubic feet). In addition, radioactive/hazardous waste associated with environmental restoration and decontamination and decommissioning activities would have a 30-year expected forecast of approximately 67,800 cubic meters (2.39 million cubic feet) (DOE 2000). During this same time period, other reasonably foreseeable activities that were not included in the 30-year forecast would add, approximately, an additional 191,400 cubic meters (6.76 million cubic feet). Therefore, the potential cumulative amount of waste generated from SRS activities during the period of interest would be approximately 558,350 cubic meters (19.72 million cubic feet). It is important to note that the quantities of waste generated are not equivalent to the amounts that will require disposal. At SRS, high-level radioactive material is evaporated and concentrated to a smaller volume for final disposal. Combustible low-level radioactive waste is volume-reduced on site in the Consolidated Incineration Facility.

The Three Rivers Solid Waste Authority Regional Waste Management Center at SRS accepts nonhazardous and nonradioactive solid waste from SRS and eight surrounding South Carolina counties. This municipal solid waste landfill provides state-of-the-art Subtitle D (nonhazardous) facilities for land-filling solid waste while reducing the environmental consequences associated with construction and operation of multiple county-level facilities. It was designed to accommodate combined SRS and county solid waste disposal needs for at least 20 years, with a projected maximum operational life of 45 to 60 years (DOE 2000). The landfill is designed to handle an average of 1,000 tons per day and a maximum of 2,000 tons per day of municipal solid waste. The SRS and eight cooperating counties had a combined generation rate of 900 tons per day in 1995. The Three Rivers Solid Waste Authority Regional Waste Management Center opened in mid-1998.

Activities supporting the treatment and management of sodium-bonded spent nuclear fuel and other planned SRS activities would not generate larger volumes of radioactive, hazardous, or solid waste beyond the current and projected capacities of SRS waste storage and/or management facilities.

4.11.3 Transportation

The Programmatic Spent Nuclear Fuel EIS (DOE 1995a) analyzed the cumulative impacts of all transportation of radioactive materials, including impacts from reasonably foreseeable future actions that include transportation of radioactive material for a specific purpose and general radioactive materials transportation that is not related to a particular action. The total worker and general population collective doses are expected to be less than 1 person-rem. The impacts of this program are quite small compared with overall transportation impacts. Total collective worker doses from all types of shipments (historical, the alternatives, reasonably foreseeable actions, and general transportation) were estimated to be 320,000 person-rem (130 latent cancer fatalities) for the period 1943 through 2035 (93 years).

4.12 PROGRAMMATIC CONSIDERATIONS

Programmatic considerations presented in this section provide information on the regulatory environment applicable to spent nuclear fuel and high-level radioactive waste. Also presented are schedule considerations for the disposal of DOE-owned spent nuclear fuel or high-level radioactive waste in a geologic repository.

4.12.1 Regulatory Environment Considerations

Prior to the acceptance of spent nuclear fuel or high-level radioactive waste at a geologic repository, certain NRC and EPA regulatory requirements and DOE Office of Civilian Radioactive Waste Management requirements must be met. Regulatory requirements specific to DOE's sodium-bonded spent nuclear fuel, are identified in the Civilian Radioactive Waste Management Office's current April 1999, Draft Waste Acceptance System Requirements Document (DOE 1999c).

One of the key NRC requirements for acceptance of spent nuclear fuel or high-level radioactive waste is that it cannot contain or generate materials that are explosive, pyrophoric, or chemically reactive (in the repository environment) in a form or amount that could compromise the repository's ability to perform its waste isolation function or to satisfy its performance objective [10 CFR 60.135(b)(1)]. The No Action Alternative may not satisfy this requirement, because the metallic sodium is highly reactive. The metallic uranium is also reactive and potentially pyrophoric, and in some cases the fuel contains highly enriched uranium, which would require criticality control measures. It also is uncertain whether the treatment technology, identified for the blanket spent nuclear fuel under Alternative 2 (cleaning the fuel to remove sodium and packaging in a high-integrity can), would be adequate to meet this NRC requirement. Under all other action alternatives, this requirement could be met.

The Waste Acceptance System Requirements Document identifies the Civilian Radioactive Waste Management Office's acceptance criteria for spent nuclear fuel and high-level radioactive waste. Under these criteria, the Civilian Radioactive Waste Management Office states that only spent nuclear fuel and high-level radioactive waste that is not subject to regulation under RCRA, Subtitle C, will be accepted for disposal. Untreated sodium-bonded spent nuclear fuel may be regulated under RCRA, Subtitle C, because it exhibits the characteristic of reactivity when exposed to water (40 CFR 261.23 (a)(2), (3)) and is ignitable (40 CFR 261.21 (a)(2)).

Under RCRA, 40 CFR 268.9 (c), "... no prohibited waste which exhibits a characteristic under 40 CFR part 261, subpart C, may be land disposed of unless the waste complies with the treatment standards under 40 CFR 268, subpart D." Deactivation is the waste treatment technology for waste that exhibits the characteristic of reactivity and ignitability (40 CFR 268.40). RCRA land disposal requirements (i.e., 40 CFR 268.40) also require generators of waste that exhibits the characteristics of reactivity to water or ignitability to identify all underlying hazardous constituents reasonably expected to be present in the waste at the point of generation, and to treat these constituents to the universal treatment standards. If the characteristic waste

is treated by the applicable treatment technology and the waste no longer exhibits the characteristic, then the waste no longer needs to be regulated under RCRA, Subtitle C, and can be managed as a nonhazardous waste (62 *FR* 62083).

The direct disposal option of the No Action Alternative may not satisfy this requirement, because the sodium-bonded spent nuclear fuel could be considered to be reactive and ignitable, and therefore, it may not be accepted for disposal at a geologic repository. All of the alternatives under the proposed action would be able to deactivate the sodium-bonded fuel and remove the characteristics of reactivity and ignitability. The metallic uranium is considered to be reactive, and in some cases pyrophoric; however, it would not be a RCRA hazardous waste because it is defined under the Atomic Energy Act of 1954, as amended (42 U.S.C. 2001 *et seq*.), as a source, special nuclear, or byproduct material and, therefore, is excluded from RCRA under 40 CFR 261.4 (a)(4).

The Waste Acceptance System Requirements Document also identifies specific acceptance criteria for DOE's spent nuclear fuel and high-level radioactive waste. For high-level radioactive waste, the Civilian Radioactive Waste Management Office specifies borosilicate glass as a standard vitrified high-level radioactive waste form. For DOE's spent nuclear fuel, specific acceptance criteria have been developed for canistered DOE spent nuclear fuel, including naval spent nuclear fuel that is intended for disposal in the canister. Performance criteria for the ceramic high-level radioactive waste and the metallic high-level radioactive waste forms are being developed. However, no specific acceptance criteria have been developed for spent nuclear fuel that has been melted into a liquid form and then solidified. The No Action Alternative may be able to meet this requirement for the disposal canisters; however, as previously discussed, it may not meet the other waste acceptance requirements (e.g., NRC and RCRA regulations).

For Alternative 3 (blanket fuel) where the treated waste form would be a vitrified borosilicate glass, the specific acceptance criteria have been developed. However, final approval of this waste form would be contingent upon the requirements in the disposal facility's license.

For Alternative 1 (blanket and driver fuel), Alternative 2 (driver fuel), Alternative 3 (driver fuel), Alternative 4 (driver fuel), and Alternative 5 (driver fuel), performance criteria for the ceramic high-level radioactive waste and the metallic high-level radioactive waste form have been developed, but need approval. Again, final approval of this waste form would be contingent upon the requirements in the disposal facility's license.

For Alternative 2 (blanket fuel), the specific acceptance criteria for canistered spent nuclear fuel would apply and may be achieved. However, the long-term durability of the proposed overpack container has not been demonstrated or documented. Without such demonstration of extended containment, the ability of the high-integrity can concept to meet the safety standards proposed by the National Research Council is unknown (National Research Council 1998).

For Alternative 4 (blanket fuel), Alternative 5 (blanket fuel), and Alternative 6 (blanket and driver fuel), the specific acceptance criteria for conditioned spent nuclear fuel would need to be developed and approved.

4.12.2 Schedule Considerations

The schedule perspective for each of the alternatives is affected by two time frames: the proposed schedule for the construction, operation, and closure of a geologic repository, and 2035, the year by which DOE committed to remove all spent nuclear fuel from Idaho under the 1995 agreement with the State of Idaho.

The proposed schedule for the repository is discussed in the *Viability Assessment of a Repository at Yucca Mountain* (DOE 1998f). A site recommendation decision for the geologic repository is scheduled for 2001. If the site were to be subsequently authorized, a license application could be submitted in 2002. The NRC construction authorization decision could occur in 2005 at the earliest. Repository construction would begin upon receipt of this authorization. DOE must update its licensing application and submit it to the NRC before the Commission will issue a license to receive and process nuclear waste. This update is scheduled for 2008. Assuming repository construction sufficient to begin waste emplacement will take five years, the first waste emplacement at Yucca Mountain could occur in 2010. DOE would design the repository to close as early as approximately 10 years after emplacement of the last waste package, or to be kept open for at least 100 years after initiation of waste emplacement, with a reasonable expectation that the repository actually could be kept open with appropriate maintenance for 300 years after initiation of waste emplacement. The Viability Assessment (DOE 1998f) assumes a reference case in which closure of a monitored geologic repository is initiated in 2110, 100 years after initiation of waste emplacement operations.

Under the No Action Alternative, the untreated sodium-bonded spent nuclear fuel could remain in storage at the current locations until 2035. After that, it would need to be transported outside the State of Idaho and stored or treated at another DOE site. If the waste acceptance criteria are finalized by 2010 and indicate that direct disposal of the sodium-bonded spent nuclear fuel is possible, the fuel could be packaged for direct disposal well before 2035.

The treatment of the driver spent nuclear fuel using the electrometallurgical technology under Alternatives 1 through 5 could start as early as 2000 and could be completed by 2006 to 2007. If the decision to select a technology is delayed until after 2010, when waste acceptance criteria may be finalized, it would require two to three years lead time for the reactivation or installation of new equipment for the electrometallurgical treatment technology and six to seven years for the processing, for a total of approximately 10 years. The high-level radioactive waste would be ready for disposal by 2020.

The treatment of driver spent nuclear fuel only using the melt and dilute process at ANL-W could start as early as 2005 and could be completed by 2007. If installation of the necessary equipment is delayed until after 2010, the conditioned spent nuclear fuel would be ready for disposal in 2017.

The treatment of the blanket spent nuclear fuel using the electrometallurgical technology under Alternative 1 could start as early as 2000 and could be completed by 2012 or 2013. Delaying a decision until after 2010 would add 10 to 15 years, depending on the time required to reactivate or install new equipment. The process still could be completed by 2030.

The preparation of the blanket spent nuclear fuel and its placement in high-integrity cans under Alternative 2 could start in 2003. Cleaning and sodium removal activities and packaging would take approximately six years and could be completed by 2009. Delaying a decision until after 2010 would delay the completion of this effort to approximately 2020.

The treatment of blanket spent nuclear fuel using the PUREX process at SRS would not start until 2005 because the F-Canyon is committed to other missions. Once started, however, all blanket spent nuclear fuel could be processed in less than one year. The decladding and sodium removal activities at ANL-W to prepare the blanket spent nuclear fuel for transportation and processing also would not start until 2003, allowing for installation of new equipment. It is estimated that preparation activities at ANL-W for all blanket spent nuclear fuel would last approximately six years. Therefore, the overall process could be completed by approximately 2010. At this time it is not clear whether the decision to process blanket spent nuclear fuel at the F-Canyon could be delayed until after 2010 because DOE has scheduled operations for the F-Canyon until 2005. If there were a programmatic decision to close the F-Canyon after currently scheduled operations are completed, the F-Canyon would not be available.

The preparation of blanket spent nuclear fuel for the melt and dilute process at ANL-W under Alternative 4 could start in 2003, allowing time for the installation of new equipment. The melt and dilute activities could start in 2005 and could be completed in seven years (by 2012). The process would require sodium removal activities at ANL-W, which could be done in parallel. The blanket spent nuclear fuel preparation activities would start in 2003 and would require approximately six years for completion. The overall process could be completed by 2012. If a decision were delayed until after 2010, treatment would not be completed until about 2020.

The treatment of blanket spent nuclear fuel using the melt and dilute process at SRS under Alternative 5 could start after 2020 if capacity becomes available. It is estimated that the facility would be operational by 2005, but it is committed to other missions until 2035, as stated in the SRS Spent Nuclear Fuel Management Final EIS (DOE 2000). If additional capacity becomes available, treatment could start as soon as 2020. Processing of the blanket spent nuclear fuel at SRS would take approximately three years. The decladding and sodium removal activities at ANL-W that are needed to prepare the fuel could start in 2003 and could be completed by 2009, well before processing begins. Delaying a decision until 2010 would push the completion of the decladding activities to 2019, which would be well before processing could begin at SRS.

The treatment of blanket and driver spent nuclear fuel using the melt and dilute process at ANL-W under Alternative 6 could start as early as 2003 and could be completed by 2015. Delaying a decision until 2010 would push completion to approximately 2025.

Table 4–73 summarizes the dates for completing the process for each alternative, given that a decision to proceed is made in the year 2000 or the year 2010.

Table 4–73 Treatment Completion Year

Alternatives [a]	*Decision in 2000*	*Decision in 2010*
No Action (direct disposal)	Before 2035	Before 2035
Alternative 1		
Driver (only)	2006	2020
Driver and blanket	2012	2030
Alternative 2		
Driver	2006	2020
Blanket	2009	2020
Alternative 3		
Driver	2006	2020
Blanket	2010	F-Canyon may not be available
Alternative 4		
Driver	2006	2020
Blanket	2012	2020
Alternative 5		
Driver	2006	2020
Blanket	2025	2025
Alternative 6		
Driver (only)	2007	2017
Driver and blanket	2015	2025

[a] See Section 2.5 for an explanation of alternatives.

4.13 MITIGATION MEASURES

Following completion of an EIS and its associated Record of Decision, DOE is required to prepare a Mitigation Action Plan to address any mitigation commitments expressed in the Record of Decision (10 CFR

1021.331). The purpose of the Mitigation Action Plan is to explain how measures designed to mitigate adverse environmental impacts will be planned and implemented. The Mitigation Action Plan is prepared prior to DOE taking any action directed by the Record of Decision that is the subject of a mitigation commitment.

Based on analyses of the environmental consequences of the proposed action presented earlier in this chapter, no mitigation measures would be necessary since all potential environmental impacts would be small and well within applicable requirements. Each DOE site would follow installation and operational practices that would minimize any potential impacts to air and surface water quality, noise, operational and public health and safety, and accident prevention and mitigation. These practices are dictated by Federal and state licensing and permitting requirements, as described in Chapter 5.

4.14 RESOURCE COMMITMENTS

This section describes the unavoidable adverse environmental impacts that could result from the proposed action; the relationship between short-term uses of the environment and the maintenance and enhancement of long-term productivity; and irreversible and irretrievable commitments of resources. Unavoidable adverse environmental impacts are impacts that would occur after implementation of all feasible mitigation measures. The relationship between short-term uses of the environment and the maintenance and enhancement of long-term productivity addresses issues associated with the condition and maintenance of existing environmental resources used to support the proposed action and the utility of these resources after their use. Resources that would be irreversibly and irretrievably committed are those that cannot be recovered or recycled and those that are consumed or reduced to unrecoverable forms.

4.14.1 Unavoidable Adverse Environmental Impacts

Implementing any of the alternatives considered in this EIS for the treatment and management of sodium-bonded spent nuclear fuel would result in unavoidable adverse impacts to the human environment. In general, these impacts are expected to be minimal and would come from incremental impacts attributed to the operation of treatment and management facilities at ANL-W and SRS.

Operation of treatment and management facilities at ANL-W and SRS would result in unavoidable increases of radiation exposures to workers and the general public. Workers would be exposed to direct radiation and other chemicals associated with the handling and treatment of the sodium-bonded spent nuclear fuel. The incremental annual dose contribution from the treatment and management of sodium-bonded spent nuclear fuel to the maximally exposed offsite individual, general population, and workers are discussed in Sections 4.3.4, 4.4.4, 4.5.4, 4.6.4, 4.7.4, and 4.8.4.

Also unavoidable would be the generation of additional low-level transuranic and mixed radioactive waste compared to baseline generation rates, which would either be treated and stored on site at ANL-W or SRS, or transported and managed off site at appropriate waste disposal facilities. Any other waste generated during treatment and management activities would be collected at the site, treated and/or stored, and eventually removed for suitable recycling or disposal off site in accordance with applicable EPA regulations.

Operation of treatment and management facilities at ANL-W and SRS would have minimal unavoidable adverse environmental impacts to air and water quality. Air quality would be affected by increases in various chemical or radiological constituents in the routine emissions typical of facility operations at these sites. Impacts to water resources and quality also would be affected by the release of various chemical or radiological constituents in the routine effluent only from PUREX processing at SRS. Impacts to the environment associated with the normal operation of facilities at ANL-W and SRS would occur regardless

of the treatment and management of spent nuclear fuel. These routine impacts also have been addressed in various other NEPA documentation at these sites.

The alternative treatment processes would generate varying amounts of waste material that could affect storage requirements. This would be an unavoidable impact on the amount of available and anticipated storage space and the requirements of disposal facilities.

4.14.2 Relationship Between Local Short-Term Uses of the Environment and the Maintenance and Enhancement of Long-Term Productivity

Implementation of the alternatives, including the No Action Alternative, would cause short-term commitments of resources (e.g., air emissions and water discharges) and would permanently commit certain resources (e.g., dilution materials and energy). For each alternative, the short-term use of these resources would result in potential long-term benefits to the environment and the enhancement of long-term productivity by decreasing overall health risks to workers, the public, and the surrounding environment by reducing their exposure to hazardous and radioactive substances. The short-term effect on workers, the public, and the environment from the treatment of sodium-bonded spent nuclear fuel would be offset by the long-term benefits of safe, stable, secure storage of these materials.

Under the No Action Alternative, environmental resources already have been committed to the storage of spent nuclear fuel. This commitment would serve to maintain existing environmental conditions with little or no impacts to the long-term productivity of the environment. The continued storage of sodium-bonded spent nuclear fuel at ANL-W and INEEL until 2035 and the potential for its direct disposal in a geologic repository would result in less exposure to hazardous and radioactive materials for workers, the public, and the environment than would be experienced under the proposed action. Only the direct disposal of the sodium-bonded fuel in a repository would have the potential to enhance the long-term viability of the environment in Idaho.

Under the proposed action, the short-term use of environmental resources at ANL-W and SRS would be greater than for the No Action Alternative. The short-term commitment of resources would include the space required for onsite processing, the commitment of processing facilities, transportation, and other disposal resources and materials for the treatment and management of sodium-bonded spent nuclear fuel. Workers, the public, and the environment would be exposed to larger amounts of hazardous and radioactive materials over the short-term from the handling and treatment of the spent nuclear fuel, including process emissions and the handling of waste. Again, these commitments would be offset by an even greater potential for enhanced long-term viability of the environment than under the No Action Alternative.

Over the life of the proposed action, groundwater would be used at SRS to meet sanitary and process needs. After use and treatment, this water would be discharged into surface water streams. Depending on the site chosen (F- or L-Area) and the technology implemented over the short-term, the resulting increases in pollutant loadings would take advantage of the natural assimilative capacity of the receiving stream(s). However, these incremental pollutant loadings should not adversely affect either short- or long-term viability of the aquatic ecosystem. These impacts would be assessed during the regulatory permitting process once an alternative has been selected.

Regardless of location, air emissions associated with implementation of any of the technologies would add small amounts of radiological and nonradiological constituents to the air of the regions around ANL-W and SRS. During the project's life, these emissions would result in additional loading and exposure, but would not impact compliance with air quality or radiation exposure standards at either site. There would be no significant residual environmental effects to long-term environmental viability.

The management and disposal of sanitary solid waste and nonrecyclable radiological waste over the project's life would require energy and space at ANL-W and SRS treatment, storage, or disposal facilities. The land required to meet the solid waste needs would require a long-term commitment of terrestrial resources. Upon the facilities' closures, DOE could decontaminate and decommission the facilities and/or equipment and restore them to brown field sites which could be available for future commercial or industrial development.

Regardless of location, continued employment, expenditures, and tax revenues generated during the implementation of any of the alternatives would directly benefit the local, regional, and state economies over the short-term. Long-term economic productivity could be facilitated by local governments investing project-generated tax revenues into infrastructure and other required services.

The use of short-term resources to operate spent nuclear fuel treatment and management facilities at either ANL-W or SRS would not affect the long-term productivity of these sites.

4.14.3 Irreversible and Irretrievable Commitments of Resources

Irreversible and irretrievable commitments of resources for each alternative potentially would include mineral resources during the life of the project and energy used in treating the waste. The commitment of capital, energy, labor, and material during the implementation of the alternatives generally would be irreversible.

Energy expended would be in the form of fuel for equipment and vehicles, electricity for facility operations, and human labor. The energy consumption of treatment and management facilities would be a small fraction of the total energy used at each DOE site. None of the technologies evaluated in the EIS would require significantly higher or lower energy consumption. Assuming that these facilities are totally dedicated to the treatment and management of sodium-bonded spent nuclear fuel, it is estimated that total electrical energy consumption would range from 101,500 megawatt hours for Alternative 2, high-integrity cans, to 130,000 megawatt hours for Alternative 5, melt and dilute at SRS. Operation of any proposed facility would generate nonrecyclable waste streams, such as radiological and nonradiological solid waste and some process wastewaters. However, certain materials and equipment used during operation of the proposed facility could be recycled when the facility is decontaminated and decommissioned.

The implementation of the alternatives considered in this EIS, including the No Action Alternative, would require water, electricity, steam, and diesel fuel. Water at SRS and ANL-W would be obtained from onsite groundwater sources and steam from existing onsite sources. Electricity and diesel fuel would be purchased from commercial sources. These commodities are readily available and the amounts required would not have an appreciable impact on available supplies or capacities. From a materials and energy resource commitment perspective, electrometallurgical treatment and PUREX process technologies would recover low-enriched uranium, which is usable as commercial reactor fuel.

The disposal of hazardous and/or radioactive waste also would cause irreversible and irretrievable commitments of land, mineral, and energy resources. Hazardous waste and low-level radioactive waste disposal would irreversibly and irretrievably commit land for its disposal. For each of the alternatives analyzed in this document, the No Action Alternative would have the least commitment of land, mineral, and energy resources.

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